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ARGONNE NATIONAL LABORATORY

IDAHO DIVISION

REPORT OF EBR-II OPERATIONS

July 1, 1966, through September 30, 1966

ARGONNE NATIONAL LABORATORY

IDAHO DIVISION

IDAHO FALLS, IDAHO

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B. Chronology of Principal Events (continued)

Date	Event
8/30	Steam generator drained and dry lay-up procedures instituted.
8/30 to 9/4	Shutdown cooler heat removal tests conducted at 700°F and 650°F. Results show heat removal of 69.5 kW with louvers closed and 366 kW with louvers open.
9/10	TREAT 13.8 kV line de-energized for connection of ZPPR construction power line.
9/13	Conducted periodic test of bus tie circuit breakers and emergency power system.
9/19	Removed control rod No. 8 and commenced gauging for installation of new rotary oscillator rod. Started primary sodium cool-down to 600°F.
9/20	Completed cool-down to 600°F. Also completed gauging for new oscillator rod.
9/28	Continued cool-down from 600°F toward 350°F.
9/30	Plant Status: Plant shutdown for maintenance and modification Primary system cool-down in progress Primary sodium temperature 398°F Secondary System drained to storage tank Primary tank heaters off and shutdown coolers open.

C. Plant Performance

1. Power

Power production in July and August, 1966, is presented in Figs. 1 through 6.

Detailed power production data are provided in Tables I and II.

2. Systems and Components

Scrams from power level of 1 MW or greater are summarized in Table III.

Primary sodium flow and pump performance are represented in Figs. 7 through 12. Similar information for the secondary sodium pump is given in Figs. 13 and 14.

Steam header temperature and pressure are plotted in Figs. 15 and 16, principally for future reference with regard to water treatment, steam generation, etc.

TABLE I.

TABLE I

OPERATING HISTORY DATA

July, 1966

Date	Reactor Critical Time	Cumulative Critical Time	Gross Thermal Energy	Cumulative Gross Thermal Energy	Gross Electrical Energy	Cumulative Gross Electrical Energy	Generator on Time	Cumulative Generator on Time	Thermal Power Range Max.	Thermal Power Range Min.
	Hrs	Hrs	MWht	MWht	MWhe	MWhe	Hrs	Hrs	MW	MW
1	0	6222.5	0	205703	0	57672	0	4438.0	0	0
2	0	6222.5	0	205703	0	57672	0	4438.0	0	0
3	0	6222.5	0	205703	0	57672	0	4438.0	0	0
4	0	6222.5	0	205703	0	57672	0	4438.0	0	0
5	0	6222.5	0	205703	0	57672	0	4438.0	0	0
6	0	6222.5	0	205703	0	57672	0	4438.0	0	0
7	0	6222.5	0	205703	0	57672	0	4438.0	0	0
8	0	6222.5	0	205703	0	57672	0	4438.0	0	0
9	0	6222.5	0	205703	0	57672	0	4438.0	0	0
10	0	6222.5	0	205703	0	57672	0	4438.0	0	0
11	0	6222.5	0	205703	0	57672	0	4438.0	0	0
12	0	6222.5	0	205703	0	57672	0	4438.0	0	0
13	0	6222.5	0	205703	0	57672	0	4438.0	0	0
14	0	6222.5	0	205703	0	57672	0	4438.0	0	0
15	0	6222.5	0	205703	0	57672	0	4438.0	0	0
16	0	6222.5	0	205703	0	57672	0	4438.0	0	0
17	0	6222.5	0	205703	0	57672	0	4438.0	0	0
18	2	6224.5	0	205703	0	57672	0	4438.0	.01	0
19	13	6237.5	17	205720	0	57672	0	4438.0	10	0
20	24	6261.5	983	206703	0	57672	0	4438.0	45	10
21	23.7	6285.2	1065	207768	0	57672	0	4438.0	45	0
22	14.8	6300.0	512	208280	0	57672	0	4438.0	45	0
23	24.0	6324.0	909	209189	0	57672	0	4438.0	45	30
24	24.0	6348.0	1080	210269	0	57672	0	4438.0	45	45
25	24.0	6372.0	1080	211349	0	57672	0	4438.0	45	45
26	24.0	6396.0	1080	212429	0	57672	0	4438.0	45	45
27	24.0	6420.0	1060	213489	0	57672	0	4438.0	45	20
28	24.0	6444.0	1080	214569	0	57672	0	4438.0	45	45
29	24.0	6468.0	1080	215649	0	57672	0	4438.0	45	45
30	24.0	6492	1034	216683	0	57672	0	4438.0	45	35
31	23.0	6515	1002	217685	0	57672	0	4438.0	45	0

TABLE II

TABLE II

OPERATING HISTORY DATA

August, 1966

Date	Reactor Critical Time	Cumulative Critical Time	Gross Thermal Energy	Cumulative Gross Thermal Energy	Gross Electrical Energy	Cumulative Gross Electrical Energy	Generator on Time	Cumulative Generator on Time	Thermal Power Range Max.	Thermal Power Range Min.
	Hrs	Hrs	MWht	MWht	MWhe	MWhe	Hrs	Hrs	MW	MW
1	4.2	6519.2	64	217749	0	57672	0	4438.0	15	0
2	19.8	6539.0	797	218546	0	57672	0	4438.0	45	0
3	24.0	6563.0	1080	219626	0	57672	0	4438.0	45	45
4	24.0	6587.0	1080	220706	0	57672	0	4438.0	45	45
5	24.0	6611.0	1080	221786	0	57672	0	4438.0	45	45
6	16.0	6627.0	694	222480	0	57672	0	4438.0	45	0
7	1.0	6628.0	0	222480	0	57672	0	4438.0	50 KW	0
8	0	6628.0	0	222480	0	57672	0	4438.0	0	0
9	0	6628.0	0	222480	0	57672	0	4438.0	0	0
10	0	6628.0	0	222480	0	57672	0	4438.0	0	0
11	0	6628.0	0	222480	0	57672	0	4438.0	0	0
12	16.0	6644.0	254	222734	0	57672	0	4438.0	45	0
13	24.0	6668.0	1080	223814	158	57830	11.6	4449.6	45	45
14	24.0	6692.0	1080	224894	328	58158	24.0	4473.6	45	45
15	24.0	6716.0	1080	225974	330	58488	24.0	4497.6	45	45
16	24.0	6740.0	1080	227054	330	58818	24.0	4521.6	45	45
17	24.0	6764.0	1080	228134	329	59147	24.0	4545.6	45	45
18	24.0	6788.0	1080	229214	330	59477	24.0	4569.6	45	45
19	24.0	6812.0	1080	230294	330	59807	24.0	4593.6	45	45
20	24.0	6836.0	1080	231374	330	60137	24.0	4617.6	45	45
21	24.0	6860.0	1080	232454	330	60467	24.0	4641.6	45	45
22	24.0	6884.0	1080	233534	330	60797	24.0	4665.6	45	45
23	24.0	6908.0	1080	234614	330	61127	24.0	4689.6	45	45
24	24.0	6932.0	1080	235694	326	61453	24.0	4713.6	45	45
25	15.0	6947.0	643	236337	189	61642	14.0	4727.6	45	0
26	3.5	6950.5	51	236388	0	61642	0	4727.6	20	0
27	17.5	6968.0	732	237120	206	61848	16.0	4743.6	45	0
28	0	6968.0	0	237120	0	61848	0	4743.6	0	0
29	0	6968.0	0	237120	0	61848	0	4743.6	0	0
30	0	6968.0	0	237120	0	61848	0	4743.6	0	0
31	0	6968.0	0	237120	0	61848	0	4743.6	0	0

TABLE III

SUMMARY OF EBR-II SCRAMS FROM POWER

July 1 through September 30, 1966

Month	Day	Time	Power Level	Trip	Remarks
July	21	2340	45 MW	Power System	Severe voltage fluctuation on 138 KV system.
July	22	0845	45 MW	Bulk Sodium High Temp.	
July	31	0344	45 MW	Power System	Severe voltage fluctuation-lightning.
August	1	1940	1 MW	Bulk Sodium High Temp.	*Apparent instrument malfunction
August	1	2030	5 MW	Bulk Sodium High Temp.	*
August	1	2148	15 MW	Bulk Sodium Low Level	Overcooling due to loss of secondary sodium flow indication on expanded range.
August	1	2310	7 MW	Bulk Sodium Low Level	**Instrument malfunction.
August	2	0148	15 MW	Bulk Sodium Low Level	**Instrument malfunction.
August	2	0255	15 MW	Bulk Sodium Low Level	**Instrument malfunction.
August	12	1537	37 1/2 MW	Reactor Building Isolation	***
August	26	1952	16 MW	Bulk Sodium Low Level	Defective motor in secondary flow recorder would not switch from expanded range.

* Investigation showed that the trip point was set too low. Trip point was reset.

** Intermittent malfunction of the millivolt-to-current converter. The amplifier section of the converter was replaced.

*** Investigation showed that a damaged connector caused the trip. The connector was replaced.

C. Plant Performance (continued)

3. Steady State Subassembly Outlet Temperatures

The plots of subassembly outlet temperatures which are presented in this report (Figs. 22 and 23), and those which have been included in previous reports, show the outlet temperatures of four subassemblies as measured by the installed thermocouples. In all, there are 25 such thermocouples in the reactor, four of which are used in the reactor shutdown circuit. The remaining 21 thermocouples send signals to three indicator-recorder devices: a 24-point strip chart recorder, the Automatic Data Logger (ADL) typewriters; and an IBM card punch. The values plotted in this report are obtained from the punched cards. Twelve of the 24 recorded values per day per thermocouple are used to compute the mean value for each day. The computation is made using data for full power (45 MW ± 2 MW) only.

The plots for this quarter show absolute values and spread in data similar to those shown for previous quarters. On July 31 all of the mean values were high; and the high, low, and mean values for thermocouples 12E6 and 16E9 are a few degrees beyond the deviations shown in previous reports. The averages of the values recorded on the ADL typewriter are not as high, and this discrepancy is believed to be due to interaction between the strip chart recorder and the card punch.

The discontinuity in the plots on August 26 is due to a planned reactor shutdown.

Modifications to the subassembly outlet temperature instrumentation are being considered to improve accuracy of output data and reliability of operation.

4. Sodium and Argon Chemistry

a. Primary Argon

The activities of argon-41, xenon-133, and xenon-135 measured in samples of the primary argon cover gas are plotted in Figs. 17 and 18.

Samples of primary argon were analyzed by laboratory chromatograph for hydrogen and nitrogen content. Results are summarized below for each month of the quarter:

	Nitrogen (vol. %)		Hydrogen (ppm, vol.)	
	Range	Average	Range	Average
July, 1966	1.37-1.88	1.60	11-34	18
August, 1966	1.15-2.14	1.88	all less than 10	
September, 1966	1.19-2.80	1.77	all less than 10	

An "on-line" gas chromatograph for future continuous analysis of hydrogen and nitrogen in the primary argon was installed in September during the plant maintenance shutdown.

b. Primary Sodium

In July and August, it was not possible to obtain reasonable plugging temperature data from the primary plugging loop. In September, the throttle valve, plugging valve and portions of the loop tubing were changed. Changes were made also in the arrangements of electrical heaters and insulation on the loop. Thereafter, the effects of operating variables were studied and a set of operating conditions was selected which promoted more sensitive plugging indication. The following plugging temperatures were measured in September:

Date	Time	Plugging Temperature (^{O}F)
9/24/66	2130	280
9/25/66	1730	282
9/26/66	0100	275
9/26/66	not recorded	270
9/26/66	1200	315
9/26/66	1800	285
9/27/66	0100	272
9/27/66	1700	285
9/28/66	0100	280
9/29/66	1415	296
9/30/66	0930	282

Samples of EBR-II primary sodium were analyzed for hydrogen and oxygen content in the Chemistry Division at ANL-Illinois. Results are tabulated below:

	Duplicate Determinations	
Date Sampled	Hydrogen* (ppm,wt.)	Oxygen** (ppm,wt.)
7/27/66	2.3, 4.0	7, 8
8/9/66	2.6, 1.6	14, 9

* By the method described in:

Holt, B.D., Determination of Hydrogen in Alkali Metals by Isotope Dilution Method; Anal. Chem., 31, p. 51. (January, 1959).

** By the method described in:

Holt, B. D., Glovebox Amalgamation for the Determination of Oxygen in Alkali Metals ANL-7123 (December, 1965).

Analysis results for samples taken in September are not available at this writing. They will be included in the next quarterly report.

c. Secondary Argon

In July and August, the continuous gas chromatograph, analyzing a sample stream from the secondary surge tank, indicated hydrogen concentrations less than 5 ppm $\left(\frac{v}{v}\right)$ and nitrogen concentrations in the range 1000 to 1200 ppm during steady-state power operation.

d. Secondary Sodium

Secondary sodium plugging temperatures and purification system operation are presented in Figs. 19 through 21.

Samples of sodium were analyzed for hydrogen by the isotopic dilution technique in the Chemistry Division at ANL-Illinois. Results were as follows:

Date Sampled	Duplicate Determinations Hydrogen (ppm,wt.)
7/20/66	3.8, 4.8
8/5/66	3.1, 5.7

5. Water Treatment and Related Work

a. Power Cycle Streams

Data are tabulated below for power operation and for hot standby with feedwater and blowdown flow:

	pH		Hydrazine (ppm)		Ammonia (ppm)	
Stream	Range	Average	Range	Average	Range	Average
Feedwater	9.0-9.9	9.57	0.01-0.20	0.10	0.6-1.6	1.16
Condensate	9.0-10.1	9.46			0.6-1.8	1.25
Blowdown	8.8-9.35	9.16	0.1-0.3	0.22	0.3-0.8	0.58
Blowdown Demineralizer Effluent	7.45-8.3	7.80	U*		U*	
Steam	9.0-9.6	9.35			0.5-1.6	1.17
Deaerator Effluent	9.1-9.5	9.36	0.01-0.03	0.02	0.6-1.6	1.16

* Undetectable by the analytical method used.

b. Cooling Water

Chemical treatment data for the plant cooling water system are tabulated below, covering the period July 1 through September 6, when the circulation of the water through the main condenser was stopped and the condenser was drained for inspection. The tabulated pH values and chromate concentrations were measured on "grab" samples of treated water from the system.

pH*		Chromate (ppm CrO_4)**		Approximate "Cycles of Concentration" ***	
Range	Average	Range	Average	Range	Average
5.2-7.8	6.3	5.0-18.1	11.2	1.4-4.1	2.4

* The specified control range is 5.8 to 6.5. Several low pH excursions were experienced as described below.

** The specified control range is 11 to 14 ppm CrO_4.

*** Calculated as the ratio:

$$\frac{\text{Specific Conductance of Treated Circulating Water at } 25^\circ C}{\text{Specific Conductance of Make-up Water at } 25^\circ C}$$

Difficulties were experienced with the control of pH by sulfuric acid injection. Malfunction or inadequate performance of the pH controller led to several low-pH excursions as indicated by the continuous pH recorder. The major excursions are listed below:

Date	Approximate Duration of Low-pH Condition (Hours)	Minimum pH Reached
7/3/66	5	4.5
7/5/66	1	5.0
7/18/66	6	4.2
7/19/66	4	4.2
7/20/66	3 1/2	5.3
7/21/66	3	5.3
7/23/66	3	4.5
7/26/66	4	4.7
7/27/66	3	4.1
7/29/66	2	5.2
7/31/66	2	5.2
8/2/66	5	3.8
8/3/66	3 1/2	5.6

8/4/66	2	4.5
8/5/66	1	4.7
8/6/66	2	4.8
8/16/66	1	4.8
8/18/66	1 1/2	5.6

These excursions were not of sufficient duration to cause extensive acidic corrosion of the system metals. However, they do point up the need for improved pH control. Until an adequate automatic control system can be provided, manual control of acid injection will be used, with close surveillance of circulating water pH by operating personnel.

c. Steam Generator Cool-Down and Dry Lay-Up

During circulation of water for cooldown of the steam generator and the power plant systems during the period August 27-29, the hydrazine injection pump was operated at full stroke to maintain a large hydrazine residual in the circulating water for scavenging oxygen. This precludes the entry of dissolved oxygen into the steam generator in case the deaerating efficiency of Feedwater Heater No. 2 is reduced during cooldown conditions.

The steam generator was drained at ambient temperature on August 30. The steam drum manholes were opened and air flow was started through the evaporator and superheater units individually to promote drying of the surfaces and to sweep out water vapor. Filtered compressed air was introduced through the fill-and-drain manifold; it left the steam generator through the open steam drum manholes. The air-drying operation was continued until September 10.

d. Inspection of Cooling System Components

(1) Steam side of main condenser

The tubes appear to be in excellent condition. The tube surface accessible to visual inspection shows no evidence of gross corrosion.

(2) Cooling water side of main condenser

The water boxes appear to be in good condition, with an adherent deposited film from the corrosion inhibitor on all surfaces. Some minor tuberculation was observed in corners not swept by water flow.

(3) Cooling water side of blowdown aftercooler

Extensive tuberculation of the steel tubesheet around the admiralty metal tubes was observed. However, they appear to be old tubercles; active corrosion does not seem to be occurring. After cleaning, the tubesheet showed some shallow pitting.

The tubes appeared to be in excellent condition, bearing a thin adherent film from the corrosion inhibitor.

(4) Cooling water sides of lube oil coolers of turbine-generator

Extensive tuberculation of the cast iron or steel heads was observed. After cleaning, the surfaces exhibited moderate to severe pitting. This corrosion is believed to result from the relatively low water flow (limiting replenishment of corrosion inhibitor), the dissimilar metals (iron and copper bearing alloys) in contact, and the relatively high water temperatures in these coolers. Magnesium sacrificial anodes will be installed in the cooler heads at the next opportune shutdown to provide cathodic protection of the steel.

The tube sheet and tubes (admiralty or Monel) were in good condition, bearing a thin film from the corrosion inhibitor.

(5) Cooling water side of Dowtherm cooler (sodium boiler building)

Minor tuberculation of the steel shell, baffles, and tie rods was observed. The copper alloy tubes are in good condition.

II. Fuel Handling

All reactor loading changes have been programmed on the basis of 1.2 a/o maximum burnup for the Mark IA fuel.

Two experimental subassemblies (X013, X014) containing structural materials were installed in the reactor prior to Run No. 20. One containing oxide fuels (X012) was installed prior to Run No. 21. Two alpha measurement subassemblies which were irradiated in Row 9 (U1548X) and in Row 11 (U1549X) were removed from the reactor following Run No. 21. The irradiation of experimental subassembly X013 was completed and it was removed from the reactor at the end of Run No. 21. Subassemblies X013, U1548X, and U1549X were in the storage basket at the end of the quarter.

A total of 45 spent subassemblies were transferred to the Fuel Cycle Facility for examination, disassembly, and reprocessing. Thirty-seven reprocessed subassemblies were received from the Fuel Cycle Facility for installation in the reactor.

The available inventory of subassemblies was depleted by the loading requirements for Runs No. 20 and 21. Ten subassemblies were available in inventory on July 1, 1966, and fifteen were available at the end of the quarter.

Two major changes in the reactor grid loadings were made for reactor Runs No. 20 and 21. Six foil-bearing subassemblies were installed for low power flux measurements (designated Run No. 20A) preceding Run No. 20 and were removed prior to starting the run. These foil subassemblies were also installed for flux measurements (designated Run No. 20B) following Run No. 20, and were subsequently removed. The core loadings for Runs No. 20 and 21 are summarized below:

Reactor Power Run	Core Size	Experimental Irradiation Subassemblies (see Figs. 22 & 23)
20	80	8 in core, 4 in inner blanket
21	80	9 in core, 4 in inner blanket

Tabular summaries of the fuel handling operations during this quarter follow:

Table IV

Run No. 20A
Foil Subassembly Installation

Subassy. No.	From	To
Wire-1		1A1
C-196	3B1	
Wire-2		3B1
C-229	4B2	
Wire-3		4B2
C-234	5B4	
Wire-4		5B4
A-706	6B5	
Wire-5		6B5
A-753	7B5	
Wire-6		7B5

Table V

Run No. 20A
Foil Subassembly Removal

Subassy. No.	From	To
Wire-1	1A1	
C-237		1A1
Wire-2	3B1	
C-196		3B1
Wire-3	4B2	
C-229		4B2
Wire-4	5B4	
C-234		5B4
Wire-5	6B5	
A-706		6B5
Wire-6	7B5	
A-753		7B5
B-342	6D2	
A-758		6D2

Table VI

Loading Changes for Reactor Power Run No. 20

Subassy. No.	From	To	Maximum Burnup
SO-1916		11E1	
U-1152	8E5		
SO-1916		TP	
SO-1916		8E5	
U-1152		11E1	
C-182	3B2		1.22
C-249		3B2	
C-184	3D2		1.22
C-250		3D2	
C-186	4F3		1.22
C-251		4F3	
B-316	6C4		1.12
B-334		6C4	
C-161	2D1		1.08
X014		2D1	
C-183	3C1		1.22
X013		3C1	
A-713	6B2		
B-335		6B2	
A-751	6C2		
B-336		6C2	
A-774	6C5		
B-337		6C5	
A-724	6D5		
B-338		6D5	
A-756	6E5		
B-339		6E5	
S-609	3D1		1.07
S-607		3D1	
A-719	6F5		
B-340		6F5	
A-758	6D2		
B-342		6D2	
C-237	1A1		

Table VII

Run No. 20B
Foil Subassembly Installation

Subassy No.	From	To
C-190	2E1	
Wire-1		2E1
C-197	3C2	
Wire-2		3C2
C-218	4E1	
Wire-3		4E1
C-244	5F2	
Wire-4		5F2
A-706	6B5	
Wire-5		6B5
A-714	7F5	
Wire-6		7F5
A-758	6D2	
B-342		6D2

Table VIII

Run No. 20B
Foil Subassembly Removal

Subassy No.	From	To
B-342	6D2	
A-758		6D2
Wire-6	7F5	
A-714		7F5
Wire-5	6B5	
A-706		6B5
Wire-4	5F2	
C-244		5F2
Wire-3	4E1	
C-218		4E1
Wire-2	3C2	
C-197		3C2
Wire-1	2E1	
C-190		2E1

Table IX

Loading Changes for Reactor Power Run No. 21

Subassy. No.	From	To	Maximum Burnup
C-188	2A1		1.20
C-252		2A1	
C-189	2B1		1.20
C-253		2B1	
C-190	2E1		1.19
C-254		2E1	
C-196	3B1		1.11
C-255		3B1	
C-197	3C2		1.16
C-256		3C2	
C-225	3E1		1.11
C-258		3E1	
C-198	3E2		1.16
C-259		3E2	
C-217	4D2		1.06
C-260		4D2	
C-187	5D2		1.15
C-261		5D2	
L-426	5A1		1.08
L-432		5A1	
L-427	5B3		1.08
L-434		5B3	
L-421	5C1		1.16
L-435		5C1	
L-423	5D1		1.16
L-437		5D1	
L-424	5E1		1.16
L-438		5E1	
L-425	5F1		1.16
L-440		5F1	
B-324	6B4		1.11
B-342		6B4	
B-333	6E3		1.11
B-343		6E3	
U-1548X	9B5		
U-1227		9B5	
U-1549X	11B6		
U-1006		11B6	
C-229	4B2		
X012		4B2	
C-213	4B3		1.06
C-229		4B3	
S-603	3A1		1.11
S-610		3A1	

Table X

Spent Subassemblies Transferred to FCF

Subassy. No.	Grid Position	Maximum Burnup	Date
B-310	6F5	1.15	7-14-66
B-315	6B2	0.69	8-16-66
B-316	6C4	1.12	8-17-66
B-317	6A4	1.19	9-23-66
B-318	6A2	0.96	9-14-66
B-320	6D5	1.09	9-9-66
B-321	6E4	1.21	7-13-66
B-323	6B3	1.19	9-16-66
B-324	6B4	1.11	9-9-66
B-333	6E3	1.11	9-9-66
C-154	5F4	1.21	7-15-66
C-155	4A1	1.18	8-30-66
C-161	5B2	1.08	8-15-66
C-170	4C3	1.22	8-16-66
C-175	2C1	1.22	7-9-66
C-179	2D1	1.22	7-8-66
C-182	3B2	1.22	9-8-66
C-183	3C1	1.22	8-11-66
C-184	3D2	1.22	8-1-66
C-186	4F3	2.22	7-29-66
C-187	5D2	1.15	9-8-66
C-188	2A1	1.20	8-29-66
C-189	2B1	1.20	8-29-66
C-190	2E1	1.19	9-8-66
C-193	5F2	1.21	7-5-66
C-196	3B1	1.11	8-25-66
C-197	3C2	1.16	8-19-66
C-198	3E2	1.16	8-19-66
C-199	4B1	1.20	9-16-66
C-207	5A2	1.21	7-6-66
C-208	5A4	1.21	8-22-66
C-209	5D4	1.21	7-7-66
C-210	5E4	1.21	7-23-66
C-211	5E2	1.21	7-11-66
C-213	4B3	1.06	9-7-66
C-214	5C2	1.10	9-13-66
C-216	4C1	1.20	9-14-66
C-217	4D2	1.06	8-25-66
C-218	4E1	1.20	9-19-66
C-224	5C4	1.10	9-22-66
C-225	3E1	1.11	9-7-66
C-231	3F2	1.10	9-14-66
S-603	5A1	1.11	9-10-66
S-609	3D1	1.07	9-10-66
S-611	3A1	1.21	7-1-66

Reprocessed Subassemblies Received from FCF

Subassy. No.	Date
B-341	9-26-66
B-347	9-12-66
B-348	9-12-66
B-349	9-15-66
B-350	9-15-66
B-351	9-16-66
B-353	9-22-66
C-249	7-13-66
C-250	7-14-66
C-251	7-15-66
C-252	7-15-66
C-253	7-28-66
C-254	8-1-66
C-255	8-2-66
C-256	8-2-66
C-258	8-4-66
C-259	8-5-66
C-260	8-5-66
C-261	8-5-66
C-263	8-8-66
C-264	8-12-66
C-266	8-15-66
C-267	8-16-66
C-268	8-20-66
C-269	8-20-66
C-270	8-27-66
C-271	8-27-66
C-272	8-27-66
C-273	9-2-66
C-274	9-2-66
C-275	9-2-66
L-437	7-26-66
L-438	7-26-66
L-440	7-28-66
L-441	7-28-66
S-607	7-1-66
S-612	7-1-66

III. Reactor Physics

A. Routine Measurements

This work covers the routine physics measurements which are involved in reactor operation. For each reactor run it is necessary to determine the reactivity change, if any, of the reactor. Accordingly, measurements are made of the power coefficient, the fuel burnup rate and the worths of control and safety rods. It is also necessary to establish limits for the next run. This reporting period includes power runs. No. 20 and No. 21 and the foil irradiation runs No. 20A and No. 20B. The pertinent reactor variables for the two runs are given in Table XI.

Table XI

Reactor Variables for Runs No. 20A, 20, 20B, 21

		20A*	20	20B*	21
Excess Reactor Reactivity					
(inhours)	Initial-	196	308	128	241
	Final-	196	196	128	143
Control Rod Banked Positions					
(in.)	Initial-	14	11.2	14	11.2
	Final-	14	11.5	14	13.0
Controlling Rod Position					
	Initial-	9.6***	4.7**	4.1***	9.3
	Final-	9.6	11.8	4.1	7.6
Overall Power Coefficient (Ih/MW)		--	1.58	--	1.62
Rod Drop Test Performed		No	No	No	Yes
Power Level (MW)		0.1	45	0.05	45
Energy Increment (MWd)		~ 0	699	~ 0	610

* Runs No. 20A and 20B were short low-power runs to irradiate flux wires for fission rate distribution measurements.

** A second controlling rod was initially at 9.7" for compensation of the loading reactivity, and during the course of the run was moved to the banked position.

*** One control rod was at 0" (fuel-bearing section completely out of core) during these runs, as required for the experimental procedure.

There are 12 control rod positions in the reactor, of which only one ordinarily is used as a shim rod (controlling rod). The usual procedure is to bank all but one of the fuel rods at some position greater than 11.2 in. and use one rod for controlling. The bank position is raised during the run so that the controlling rod operates in the range of 5 in. to 11 in. During runs 20 and 21 a special low-reactivity stainless steel rod occupied one of the control rod positions. This stainless steel rod, used for the "rod drop" test of reactor stability, is fully inserted (14 in.) during steady power operation.

Each new control or safety rod inserted into the reactor grid is calibrated either by comparison with a rod of known worth or by means of incremental reactor period measurements. Between Run No. 20 and Run No. 21 a number of Mark I type control rods were replaced with Mark IA types, and the remaining Mark I safety rod was replaced with a Mark IA rod. Calibration data for the rods are given in Table XII.

Table XII

Control and Safety Rod Calibrations

Rod No.	Total Worth (Inhours) Run No. 20	Run No. 21
1	127	139*
2	146	146
3	123	138*
4	156	156
5	136	145*
6	158	158
7	144	137
8	158	158
9	Stainless Steel Rod	Stainless Steel Rod
10	158	162*
11	127	133*
12	161*	161*
Safety Rods	1.2% $\Delta K/K$	

* Mark IA fuel elements

B. Experiments

1. Neutron Source

During Runs No. 20 and 21 a new type of neutron source was resident in position 8E5. This source contains antimony oxide (Sb_2O_4) in powder form, canned in stainless steel. After removal it was to be used as a neutron source for developing the techniques of neutron radiography. The installation of the Sb2O4 source was also expected to give some information on the activation level and the source effectiveness in a Row 8 position.

After Run No. 21, during fuel handling operations, the counting rates were determined for several different positions of the new source and the regular source (antimony metal clad with tantalum) in position 7E2. The following results were obtained:

	Counts per 3 min.
Both sources in reactor	28198
Sb_2O_4 source in; regular source removed from 7E2	6061
Sb_2O_4 source out; regular source in 7E2	22709
Both sources out	342
in 8E5	16948

Calculations showed that the regular source was at about 40% of saturation and the Sb_2O_4 was at about 20% of saturation. From these experiments three conclusions can be reached:

a. The Sb_2O_4 source in Row 8 would probably be adequate for a startup source.

b. On an atom basis, the Sb_2O_4 source canned in stainless is a more efficient source than the Sb metal canned in Ta.

c. There is not enough high energy radioactive background to produce sufficient neutrons in the Be thimble without an activated gamma source.

2. Flux Wire Results

Prior to and following Run No. 20, U-235 fission rate measurements were made at selected positions in the core using the special wire subassemblies. In the first series (Run No. 20A) an attempt was made to position the wires in a "clean region" of the core, and in the second series (Run No. 20B) the wires were placed in the vicinity of experiments. A sixth row position was duplicated in each run for normalization. Figure 24 shows the loading of the core for Run No. 20, with the locations of the flux wires. Figure 25 shows the radial fission rate distributions for the two runs, together with the calculated distribution for an homogenized core approximating the Run No. 20 loading. The calculation used the SNARG code (1 dimension transport, 22 energy groups). The foil data show the real effect of localized perturbations due to the heterogeneity of the loading. They emphasize the need for additional capability to measure fission rates throughout the reactor and in the vicinity of experiments. It has not been possible to characterize the measured fission rate shapes by any single computer code, though the SNARG code seems to agree quite well with the average measured values.

Since the fission rate distributions are essential for measuring burnup in the core and power density in experiments, it is imperative that continued effort be expended in this work. Special subassemblies are being fabricated which will allow a more complete program of measuring fission rates on a continuing basis.

3. Rod Drop Test

Rod drop data were taken in Run No. 21. An analysis of the data showed no significant change in the magnitude of the prompt feedback. The control rod used for these measurements will be removed and replaced with the rotating oscillator rod.

C. Burnup Calculations

All burnup calculations are based on the following definition: Burnup is the ratio of heavy atoms ($A>232$) fissioned to the total number of heavy atoms initially present, where the ratio is averaged over some specific volume and is expressed in percent.

For the purposes of EBR-II, two burnup values are calculated: the maximum burnup in the subassembly (i.e., centerline value for the pin closest to the reactor center), or the average subassembly burnup, with reference to the burnup averaged over the length of the fuel pin and over all the fuel elements in the subassembly.

A computer program has been written to keep a running account of the maximum and average burnup of every subassembly in the core and inner blanket during its residence in the reactor. The calculation is made for each power run, taking into account changes in loading, reactor power, and the total length of the run. Based upon 200 MeV/fission and the power level as indicated by the reactor heat balance, the total number of fissions occurring during the run is calculated. These total fissions are then apportioned over the reactor according to the fission rate distribution (axial and radial) and the relative amounts of U-235, U-238, and Pu-239 in each subassembly. The burnup for that run is then added to the previous accumulated burnup for each subassembly. A running record is also kept of the Pu production and the U-236 production. As a further aid in fuel management, the computer prints out a listing of the MWd of power operation permissible for each subassembly before it reaches its prescribed maximum burnup level.

Obviously such a method is dependent upon the accuracy of the fission rate distribution for the three fissionable isotopes, upon the somewhat arbitrary choice of 200 MeV/fission, and upon the precision of measuring MWd of operation. Since the last two terms remain constant from run to run, they affect the absolute burnup calculation but not the relative burnup, the latter being the real purpose of the code. The relative fission rates, however, are very definitely a function of loading and ideally should be adjusted after every loading change. Because complete codes capable of predicting the effects of small changes in core loading have not been available, the flux distribution for the reference reactor (Hazards Summary Report, EBR-II, ANL-5719) has been used with slight modifications for axial maximum-to-average fission rates. Recent flux measurements indicate that the fission rate distributions have changed significantly. As noted above, further experimental work is planned to better define the fission rate distribution.

D. Theoretical Calculations

A modest program of reactor calculations is in progress. The codes used to date are called MACH I, CANDID and SNARG. All three codes are based on a 22-group cross-section set (Number 224) developed for ZPR-3 calculations. MACH I is a one-dimensional diffusion theory code which can be used in either cylindrical or slab geometry. CANDID is a two-dimensional diffusion theory code which is used in r,z; r,θ; or x,y geometry. The 22-group set is reduced to 6 groups for this calculation. The SNARG code utilizes transport theory in one dimension. Each code has certain advantages and disadvantages, and the attempt is being made to determine how best to characterize the reactor, in terms of its changing shape, so as to best utilize the potential of the computer programs.

A supplementary program of reactor physics calculations is in progress in the Reactor Physics Division at ANL-Illinois. This program, coordinated with the ID work, is aimed at providing an improved understanding of EBR-II neutronics with the goal of making accurate predictions of the changes in neutronic variables which result from reactor loading changes.

IV. Experimental Irradiations

A. Experimental Subassembly Locations

Figures 26 and 27 show the locations of all experimental subassemblies in the grid during reactor power Runs No. 20 and 21 as well as the locations of other special subassemblies, control and safety rods and standard EBR-II driver subassemblies.

B. Experimental Subassembly Contents and Exposure Status

Descriptions of experimental capsules and exposures in the experimental subassemblies resident in the reactor during the report period are given in Table XIII.

V. Systems Maintenance, Improvements and Tests

A. Electrical

The high voltage (2.4 kV and 13.8 kV) circuit breakers are scheduled for preventive maintenance on an annual basis. The work includes the following:

1. General inspection and cleaning of the breaker and the cubicle.
2. Check of contacts for pitting, burring, alignment, lubrication, etc.
3. Actual check of operation, mechanical and electrical. This test includes a determination of minimum trip voltage.
4. Check of all connectors and bolts for tightness, etc.

Following is a list of the high voltage breakers on which the above maintenance was performed during the report quarter:

2.4 kV Unit No. 1, Primary Pump No. 1 Feeder
2.4 kV Unit No. 2, Primary Pump No. 2 Feeder
2.4 kV Unit No. 3, 2.4 kV Incoming Line No. 1
2.4 kV Unit No. 4, Pump House No. 1 Feeder
2.4 kV Unit No. 5, Pump House No. 2 Feeder
2.4 kV Unit No. 6, Secondary Sodium Pump
2.4 kV Unit No. 7, Circulating Water Pump No. 1
2.4 kV Unit No. 8, Circulating Water Pump No. 2
2.4 kV Unit No. 9, 2.4 kV Incoming Line No. 2
2.4 kV Unit No. 10, Induction & Resistance Heating
2.4 kV Unit No. 11, Motor Driven Feedwater Pump
13.8 kV Unit No. 1, TREAT feeder.

Protective relays associated with the high voltage circuit breakers are tested on a biannual schedule. Tests are performed to verify actual operation within prescribed time limits as specified by the Sargeant & Lundy Report

No. SL 1768. The following relays were tested during the report period:

Quantity	
14	Thermal overcurrent
1	Instantaneous overcurrent
11	Time overcurrent (induction)
1	dc timing relay
3	Directional overcurrent
3	Voltage
3	Frequency

Preventive maintenance and testing were performed on 20 low-voltage (480 V), power circuit breakers in the normal and emergency switchgear. The bus-tie breaker for MCC-R1 was also removed, repaired and returned to service after failure during a test.

Tests on these breakers include high current trips at 300% and 100% of the breaker coil ratings. The 400 KW diesel generator output breaker was replaced by a new spare breaker because of broken insulators and pitted contacts. The MCC-R1 bus-tie breaker failed to close during a simulated power failure and the closing relay and closing coil burned out. The closing linkage was found to be disconnected, probably caused by vibration.

Individual motor circuits are also tested on an annual schedule. The tests include actual trip of the supply breaker at 300% breaker rating (where possible), trip of motor overloads at 400% motor full load current, and insulation resistance measurements (with a 500 V dc megger) of the motor, feeder cables and control circuit. A total of 41 motor circuits was completed this quarter. These included the condensate pump, startup feedwater pump, cooling tower fans, thimble cooling compressors and air supply fans for the reactor building.

B. Mechanical

1. Generator Inspection and Repair

During the quarter both the turbine and generator were inspected, with a vendor's representative present. Prior planning had called for concurrent inspection of these units during the major plant shutdown scheduled for late summer. However, it became necessary to inspect the generator in June when, during a routine overspeed test, a loud noise was heard followed by noticeably increased vibration.

The subsequent inspection disclosed that one of the two "cleats" (covers mounted over the field lead connectors) had fractured through its mounting flanges and had been thrown off the field. It struck several armature end turns, causing some damage to their insulation.

The vibration that resulted from this unbalance may have caused slight damage to one of the generator bearings. However, the inspection revealed that there was a slight misalignment of generator and turbine shafts, which could also have caused the bearing damage.

It was necessary to replace the two field "cleats", repair the damage to the armature and replace the damaged bearing. Alignment of the generator and turbine was corrected, the machine was reassembled after completion of the inspection and was put back into service. Its operation is now satisfactory.

2. Turbine Inspection and Repair

The inspection of the turbine was carried out in August as planned. After the casing was opened, it was discovered that a spill strip had broken loose and damaged several rows of blading. It therefore became necessary to remove the rotor, replace one row of blading and straighten the rest. All components and accessories were inspected and any necessary work was performed. Alignment was checked again and machine reassembly was begun.

3. Main Steam Stop Valve Modification

Installation of a "Limitorque" motor operator on the main steam stop valve in the sodium boiler plant was completed during this quarter. The valve can now be operated from either of two push-button stations-one mounted locally and the other in the control room. Advantages gained are: (1) Reduced manpower demand because one man can operate the valve by push-button control; and (2) Plant safety and dependability are enhanced. Operating personnel can remain in the control room. In case of emergency, the valve can be closed quickly and remotely.

4. Cooling Tower Maintenance

Considerable maintenance work was performed on the cooling tower during this quarter. Two fan drive gear reducers were repaired. The input shaft bearing was replaced on one unit. The second unit was taken to the shop for repairs to gears, shaft and oil pump and replacement of the output shaft bearing. The failure was apparently due to water leakage into the unit and subsequent freezing.

Missing fill splash panels and damaged fan stack panels were replaced. Steel pipe and structural members were painted. The basin was drained and cleaned. A section of acid injection line, found to be in unsatisfactory condition, probably due to corrosion at a minor weld defect, was replaced.

5. Fuel Unloading Machine

A minor modification was made to change the operation of the Fuel Unloading Machine (FUM) during "FUM-to-Basket" and "Basket-to-FUM" operations, which is the main mode of operation with the reactor at power. In the new fuel handling sequence the FUM gripper is raised only high enough (25 ft) to clear the subassembly left on the transfer arm. The addition of a new FUM elevation switch then allows a normal transfer to the basket and a faster return of the subassembly from the basket to the FUM without requiring the FUM gripper to leave the sodium. Previously, the FUM gripper had to be at the full up position before a basket transfer could be made. The modified operation saves approximately 25 minutes per transfer and halves the number of cycles of the FUM gripper into the sodium for this type of fuel handling operation.

Local readout pressure gages were installed in the FUM Argon Cooling System.

The vapor traps and the gas purifier for the FUM Argon Cooling System were replaced because of suspected plugging. A new orifice flow meter

was installed to obtain a more accurate flow measurement in the system. The vapor trap on the outlet line from the primary tank contained about 20 lbs of sodium.

The primary tank transfer port guide tube was cleaned several times. Cleanup and repairs of the primary tank transfer port and the FUM transfer port was effected several times.

Repair of the FUM gripper was necessary about every week during fuel handling operations. Minor changes to prevent mechanical interference were accomplished during the extended shutdown. New types of coupling and spring motor have been ordered for testing with the goal of reducing the frequency of maintenance.

6. Reactor Building Ventilation System

Inspection plates were installed in the Reactor Building air inlet and air exhaust ducts for checking the valve seats.

7. Main Condenser Desuperheater Sprays

The large and small desuperheater sprays for the main condenser were removed, cleaned, repaired and reinstalled.

8. Emergency Air Lock

The annual leak rate test of the Reactor Building emergency air lock outer door and compartment was not satisfactory on the initial test. The electrical penetration for the lock was found to be the main source of leakage. The penetration was changed to a "Conax" seal and subsequent leak rate test results were excellent, with a leak rate of only 4.18 ft^3/day. All other leak rate tests performed on reactor building penetrations during this report period were satisfactory.

9. Steam Valve (VC-501-B)

The small bypass steam valve (VC-501-B) has been a chronic maintenance item. The valve was repaired three times during this report period. A replacement valve of improved design has been ordered.

10. Startup Feedwater Pump

The startup feedwater pump cylinders were repacked twice.

11. Reactor Building Emergency Air Compressor

The Reactor Building emergency air compressor failed. New finger valves were installed and the unit was returned to service. The output was checked and found to be as rated by the manufacturer.

12. Steam Drum Gage Glasses

Double isolation valves were installed on all of the steam drum gage glasses and gages were repaired to eliminate leakage.

13. Secondary Sodium Pump MG Set

The coupling on the secondary sodium pump MG set was disassembled for inspection to determine the source of a noise during operation. The teeth on the coupling were found to be peened rather badly and a key on the shaft of the coupling was possibly causing some interference. The key was repaired and a new coupling has been ordered.

14. Main Condenser

Three brass studs were found broken off in the water box of the main condenser. New studs have been ordered for replacement during a future maintenance shutdown period.

15. Primary Purification System

The bellows in the throttle valve and plugging valve for the primary purification system plugging loop were found to be ruptured, probably because of insufficient heating before operation. The valves were repaired and reinstalled using "Conoseal" fittings.

16. Rotating Plug Seals

Improved access was provided to the outer annuli (air sides) of both seal troughs. A new hole (separate from the heater holes) was drilled through the small plug and through the spring steel gas seal above the trough outer annulus. This hole provides direct vertical access to the alloy.

Similar vertical access to the outer annulus of the large plug seal trough was provided by enlarging the "window" heater hole and penetrating the gas seal.

Extensive cleaning operations were conducted through these new access holes for removal of "drossy" oxidized material from the troughs. These operations, and subsequent make-up alloy addition, are reported in detail in the Progress Reports for September, 1966 (ANL-7255) and October, 1966 (ANL-7267).

In addition to the trough cleaning work, other maintenance jobs were performed on the rotating plugs and their drive systems. The plug bearings were greased. Electronic components in the power supply for the small plug drive were replaced after a test had disclosed that the specified voltage was not being applied to the dc drive motor. The clutches of the plug drive assemblies were removed, cleaned, adjusted and calibrated for proper torque settings and replaced.

C. Instrumentation and Control

1. Trace Heating System Control

Work completed in this quarter comprised replacement of all final control relays with solid state switches for the 75 induction and resistance heating circuits.

The solid state switches and operators were assembled on one insulated board which functions as a heat sink and which duplicates the size of the former relay package. The substitution was made and the system check-out was satisfactorily completed. Approximately 200 operating hours for aging of components had been completed prior to placing the system into complete operation for heatup and filling of the secondary sodium system.

2. Mark II Oscillator

The Mark II oscillator control wiring was partially installed. New cables were installed between the Fuel Handling Console and the oscillator. Two new equipment and control cabinets were fabricated and installed in the cable routing room. Wiring of this equipment was started.

3. Bulk Sodium and Core Subassembly Thermocouple Instruments

The indicator-controller supplied with the initial instrumentation has been replaced with solid state millivolt to milliampere converters, monitor switches, and readout equipment. Test circuits have been added to permit a check of each of the systems with the reactor in operation. The 175°F reference temperature junctions have been replaced with 32°F reference junctions for improved system accuracies. To reduce packaging density, this equipment was mounted in a new cabinet.

4. Auxiliary Controls for Containment Vessel Isolation, Ventilation and Purge Valves

A push-button station has been installed in the main control room for Reactor Building air inlet valves (R13-VR-306 and R13-VR-318), suspect exhaust valves (R13-VR-320 and R13-VR-323A) and purge exhaust valves (R13-VR-317 and R13-VR-319). The circuit design is such that there is no interference with the normal operation of the isolation system. This modification provides control of the air inlet and suspect and purge exhaust valves for containment of internal contamination or for preventing external contamination from entering the building.

5. Secondary Sodium Recirculating Pump Trip

This modification provides an interlock in the secondary sodium recirculating pump control circuit that automatically stops either or both pumps when a partial or complete dump of secondary sodium system is initiated. Provisions have been added for initiating a partial dump from the main control room. A complete dump can be initiated either in the main control room or the Sodium Boiler Plant.

6. Primary Argon Pressure Alarm

Low and high pressure alarms have been added to the "N-2" nozzle, duplicating the alarm system on the "N-1" nozzle, as a backup in case of nozzle plugging by sodium or sodium oxide deposition.

7. Argon Cooling System Pressure Indication

A pressure indicator was added to the Argon Cooling System Control Console.

8. Flux Level Recorders

The flux level strip chart recorders for power range flux monitoring have been replaced. The previous recorders were obsolete models for which repair parts were no longer available.

9. Reactor Building Purge Air Supply Fan Motors

With the Reactor Building isolated, it was necessary for an operator to leave the control room to restart the purge fan motors. This requirement was eliminated by a wiring revision giving automatic restarting of the motors with the operation of the isolation reset push-button.

10. Radioactivity Monitor for Reactor Building Discharge Air

An "ARMS" unit was installed on the air discharge line to the stack for air sampling and monitoring. The unit is located in the Reactor Building basement with readout and alarm in the main control room.

11. Control of Condensate Return to No. 2 Feedwater Heater

The normal control valve is too large for adequate flow control during plant standby operation. A small control valve has been installed in parallel with the large valve for low range flow control. The control valves have been adjusted to operate in such a manner that the small valve is fully open when the large valve starts to control.

12. Feedwater Heater Low Level Alarms

Low level alarms have been added to each of the feedwater heaters to alarm for low water level condition.

E. Special Tests

1. Rod Drop Test

During the monthly checks of drop times for the control rods, Rod No. 12 has consistently required 60 to 100 milliseconds longer total drop time than the other control rods. To determine the cause of the time delay, control rod drives No. 12 and No. 2 were instrumented for checking time increments in the sequence of dropping the rods. The times were measured for the following events:

a. Magnetic clutch de-energized

b. Magnetic clutch release

c. Rod travel of ten inches

The following table is a summary of the times, measured in milliseconds, for each of the events occurring in sequence.

	Run No. 1		Run No. 2		Run No. 3	
	Rod 2	Rod 12	Rod 2	Rod 12	Rod 2	Rod 12
Scram Initiated	0	0	0	0	0	0
Magnetic Clutch De-energized	13.5	13.5	6.9	6.9	6.7	6.7
Magnetic Clutch Release	51	105	44	78	44	82
Rod Travel (10 in.)	175.5	195.5	175	204	173	203
Totals	239	314	225	289	228	292

From the table it can be seen that Control Rod No. 12 required 20 to 30 milliseconds more rod travel time than Control Rod No. 2 for the 10-inch drop. This is an acceptable variance between rods. The clutch of drive No. 12 required 34 to 54 milliseconds more time to release. This release time probably can be shortened by rework or replacement of the clutch assembly. This work will be scheduled in the next quarter.

2. Constant Power Supply Test

Using auxiliary and test instrumentation, the Process Constant Power Supply was paralleled with the site power system to check load sharing capabilities, frequency stability and filtering effects of the combined systems.

The test results indicate that (1) the Process Constant Power Supply can be paralleled with the site system and will maintain a base load under varying load conditions, (2) the frequency of the Constant Power Supply locks with that of site power system, and (3) the site power system acts as a filter, reducing the total harmonic distortion of the Constant Power Supply to less than three per cent.

3. Shutdown Cooler Heat Removal

The shutdown cooler tests were conducted in two steps. The first step was to determine the effect of atmospheric conditions, if discernible, on shutdown cooler performance. The second step was to determine the absolute value of heat removal capability.

Step one consisted of operating the coolers at full capacity and accumulating data on wind direction, wind velocity, ambient air temperature, exit air temperature from the heat exchanger and the flow rate of the NaK. Data were collected at 1-hr intervals for approximately seven days. The results of this test indicate no significant variation in the heat removal capability with a change in wind velocity of 35 mph and a variation in ambient air temperature of 40°F. The standard deviation in the sodium flow was 2% of average flow over the entire monitoring period. A check of the data showed no correlation in flow variation with the wind or ambient air temperature. This result is difficult to interpret. However, it may indicate that shutdown cooler performance is limited by the stack.

The absolute power measurement, step two, was accomplished by maintaining the primary system at a fixed operating temperature using electrical

heaters and collecting data on shutdown cooler NaK flow and temperature rise. All other heat addition or removal processes were maintained at constant conditions as far as practicable. Data were collected for 12 hours with the shutdown cooler dampers open and for 12 hours with the dampers closed. The major source of error in determining the heat removal capability was found to be drift in primary sodium temperature.

Test results indicate that the two shutdown coolers have a combined heat removal capability equivalent to 336 kW at 700°F primary system conditions.

4. Emergency Power System Test

A test procedure has been provided for semiannual testing of the emergency power system and bus-tie circuit breakers. The test is performed by simulating loss of power on the emergency section of MCC-R1 and -P1 in turn, and then on MCC-R2 and Emergency Switchgear E1.

The test was conducted on September 13. Difficulties were encountered in the operation of bus-tie circuit breakers MCC-R1 and MCC-P1. Necessary actions were instituted for correction and repair.

The 100 kW diesel generator functioned normally throughout the test. The time delay of the relay which co-ordinates the diesel starting cycle with the bus-tie breaker on MCC-R1 was adjusted.

The 400 kW diesel generator functioned properly during the test of September 13.

TABLE XIII
SUMMARY OF CAPSULE IRRADIATIONS IN EBR-II
9/30/66

IRRADIATED TO 9/30/66

CAPSULES, FUEL - 134
CAPSULES, MTLS. - 88
TOTAL 222
NO. OF SUBASSY'S. - 15

SUBASSEMBLY	GRID LOCATION	EXPERIMENTER(S)	GOAL EXPOSURE MWd	ACTUAL FINAL EXPOSURE MWd	DATE INSTALLED	DATE REMOVED	FUEL CAPSULES: FUEL	FUEL CAPSULES: NO. OF CAPS & DESIGNATION	POWER GENERATION kW/FT. MAX.	POWER GENERATION kW/FT. MIN.	MID-PLANE BURNUP RATES ($^a/o$/MWd) $\times 10^4$ MAX.	MID-PLANE BURNUP RATES ($^a/o$/MWd) $\times 10^4$ MIN.	STATUS AS OF (9-30-66) MWd	STATUS AS OF (9-30-66) %BU(MAX)	MATERIAL CAPSULES: NO. OF CAPS & DESIGNATION	MATERIAL	TYPE OF SAMPLES: BURST TEST	TYPE OF SAMPLES: TENSILE	TYPE OF SAMPLES: CREEP RUPTURE
XA01	6D2	ANL-MET	14,000	3,940*	5-6-65	3-24-66	U-Pu-Fz	19- C93 C97 C98 C99 C100 C101 CA01 CB02 CB03 CB04 CD01 CD02 CG02 CG03 CJ01 CM01 LA02 PA01 PB02	2.7	2.0	1.23	.91	3940	0.48					
XG01	4F2	GE	700	381*	5-6-65	5-23-65	UO_2-20PuO_2	6- F1A F1B F1C F1D F1E F1F	16	14	5.78	5.08	381	0.22					
		GE													4- P1A P1B MT1 MT2	 347 347 HAST X. INCO-625 I-800 HAST-X, INCO-625 I-800	 X X	 X X X X	
XG02	7A1	GE	13,600		7-16-65		UO_2-PuO_2	1- FOE	5.3		1.99		7379	1.47					
XG03	7D1	GE	19,450		7-16-65		UO_2-PuO_2	2- FOA FOC	5.3	4.6	1.99	1.83	7379	1.47					

TABLE XIII (CONT'D)

SUBAS-SEMBLY	GRID LOCA-TION	EXPERI-MENTER(S)	GOAL EXPOSURE MWd	ACTUAL FINAL EXPOSURE MWd	DATE INSTALLED	DATE REMOVED	FUEL CAPSULES: FUEL	FUEL CAPSULES: NO. OF CAPS & DESIG-NATION	FUEL CAPSULES: POWER GENERATION kW/FT. MAX.	POWER GENERATION kW/FT. MIN.	FUEL CAPSULES: MID-PLANE BURNUP RATES $(^a/o/MWd)\times10^4$ MAX.	MID-PLANE BURNUP RATES MIN.	FUEL CAPSULES: STATUS AS OF (9-30-66) MWd	STATUS AS OF (9-30-66) %BU(MAX)	MATERIAL CAPSULES: NO. OF CAPS & DESIG-NATION	MATERIAL CAPSULES: MATERIAL	TYPE OF SAMPLES: BURST TEST	TYPE OF SAMPLES: TENSILE	TYPE OF SAMPLES: CREEP RUPTURE
XG04	7B1	GE	39.000		7-16-65		UO_2-PuO_2	2- FOB FOD	5.3	4.6	1.99	1.83	7379	1.47					
XG05	4C2	GE	10.300		9-3-65		UO_2-PuO_2	9- F-2-C F-2-D F-2-G F-2-H F-2-O F-2-R F-2-T F-2-V F-2-X	14.6	12.8	5.78	5.08	6952	4.02	5- L-2-A L-2-C L-2-E L-2-G L-2-I	 I-800 316 L 347 304 321	 X X X X X	 X X X X X	
		ANL					UC-PuC	3- HMV-5 NMV-11 SMV-2	18.3	17.8	5.38	5.20	6952	3.74					
		ANL					U-15Pu-10Zr	2- NC-17 ND-24	8.1	8.0	4.97	4.89	6952	3.46					
XG06	4E2	GE	20,600		9-3-65		UO_2-PuO_2	12- F-2-A F-2-B F-2-E F-2-F F-2-N F-2-P F-2-Q F-2-S F-2-U F-2-W F-2-Y F-2-Z	14.6	12.8	5.78	5.08	6952	4.02	5- L-2'-K L-2'-M L-2'-O L-2'-P L-2'-Q	 I-800 316 L 347 321 304	 X X X X X	 X X X X X	
		ANL					U-15Pu-10Zr	2- NC-23 ND-23	8.7	8.1	5.32	4.97	6952	3.70					

TABLE XIII (CONT'D)

SUBASSEMBLY	GRID LOCATION	EXPERIMENTER(S)	GOAL EXPOSURE MWd	ACTUAL FINAL EXPOSURE MWd	DATE INSTALLED	DATE REMOVED	FUEL CAPSULES: FUEL	FUEL CAPSULES: NO. OF CAPS & DESIGNATION	POWER GENERATION kW/FT. MAX.	POWER GENERATION kW/FT. MIN.	MID-PLANE BURNUP RATES $(^a/o/MWd) \times 10^4$ MAX.	MID-PLANE BURNUP RATES $(^a/o/MWd) \times 10^4$ MIN.	STATUS AS OF (9-30-66) MWd	STATUS AS OF (9-30-66) %BU(MAX)	MATERIAL CAPSULES: NO. OF CAPS & DESIGNATION	MATERIAL	TYPE OF SAMPLES: BURST TEST	TYPE OF SAMPLES: TENSILE	TYPE OF SAMPLES: CREEP RUPTURE
XA07	4D3	ANL	18,600		10-27-65		U-15Pu-9Zr	16- ND-25 ND-26 ND-27 ND-28 ND-29 ND-30 ND-31 ND-32 ND-33 ND-34 ND-35 ND-37 ND-39 ND-41 ND-43 ND-44	8.9	7.8	5.48	4.81	6215	3.40	3- As-9 As-10 As-11	 V-20Ti HAST-X 304			 X X X
XA08	4F2	ANL	19,800		12-13-65		(Pu-U)C	8- HMV-1 HMV-4 HWMP-1 HWMV-1 NMP-2 NMV-4 NMV-7 NMV-12	25.0	16.2	5.83	5.10	5130	3.00	9- As-1 As-2 As-3 As-4 As-5 As-6 As-7 As-8 As-12	 V-20Ti V-20Ti HAST-X HAST-X 304 V-20Ti HAST-X 304 V-20Ti	 X X X X X		 X X X X
X009	4A2	UNC	5,130		3-24-66		PuC-UC	3- UNC-78 UNC-79 UNC-80	26.5	18.5	5.74	5.56	4310	2.47					
		ANL					PuC-UC	3- SMP-1 SMV-1 VMV-1	25.5	17.1	5.83	5.10	4310	2.51	3- As-14 As-15 As-27	 V-20Ti V-20Ti 304		 X X X	
		ANL					UO_2-PuO_2	2- SOV-5 SOV-6	14.0	13.3	5.33	5.16	4310	2.30					

TABLE XIII (CONT'D)

SUBAS-SEMBLY	GRID LOCA-TION	EXPERI-MENTER(S)	GOAL EXPOSURE MWd	ACTUAL FINAL EXPOSURE MWd	DATE INSTALLED	DATE REMOVED	FUEL CAPSULES: FUEL	FUEL CAPSULES: NO. OF CAPS & DESIG-NATION	POWER GENERATION kW/FT.: MAX.	POWER GENERATION kW/FT.: MIN.	MID-PLANE BURNUP RATES ($^a/o$/Mwd)$\times 10^4$: MAX.	MID-PLANE BURNUP RATES ($^a/o$/Mwd)$\times 10^4$: MIN.	STATUS AS OF (9-30-66): MWd	STATUS AS OF (9-30-66): %BU(MAX)	MATERIAL CAPSULES: NO. OF CAPS & DESIG-NATION	MATERIAL CAPSULES: MATERIAL	TYPE OF SAMPLES: BURST TEST	TYPE OF SAMPLES: TENSILE	TYPE OF SAMPLES: CREEP RUPTURE
X009 (CONT'D)		PNWL(ANL)					PuO_2-S/S	2- 5P-13 5P-14	9.6	6.1	6.19	6.19	4310	2.66	6- A-1 A-2 A-5 A-6 L-4-C L-4-D	 304 304 304 304 316 316	 X X	 X X X X X	 X X X X
X010	7F3	GE	19,600		3-24-66		UO_2-PuO_2	4- FOJ FOK FOL FOM	8.1	7.3	3.00	2.77	4310	1.29					
		ANL													5- As-16 As-17 As-18 As-19 As-20	 V-20Ti V-20Ti V-20Ti HAST-X V-20Ti, 304 HAST-X		 X X X	 X X
		PNWL													10- A-3 A-4 A-7 A-8 As-21 As-22 As-23 As-24 As-25 As-26	 304 304 304 304 V-20Ti 304 304 304 304 304	 X	 X X X X	 X X X X
X011	2F1	ANL	8,300		5-9-66		UO_2-20P_uO_2	7- HOV-4 HOV-10 HOV-15 SOV-1 SOV-3 SOV-7 TVOV-1	19.2	16.2	6.11	5.81	2739	1.67					

TABLE XIII (CONT'D)

SUBAS-SEMBLY	GRID LOCA-TION	EXPERI-MENTER(S)	GOAL EXPOSURE MWd	ACTUAL FINAL EXPOSURE MWd	DATE INSTALLED	DATE REMOVED	FUEL CAPSULES: FUEL	FUEL CAPSULES: NO. OF CAPS & DESIG-NATION	POWER GENERATION kW/FT.: MAX.	POWER GENERATION kW/FT.: MIN.	MID-PLANE BURNUP RATES ($^a/_o$/MWd)×10^4: MAX.	MID-PLANE BURNUP RATES ($^a/_o$/MWd)×10^4: MIN.	STATUS AS OF (9-30-66): MWd	STATUS AS OF (9-30-66): %BU(MAX)	MATERIAL CAPSULES: NO. OF CAPS & DESIG-NATION	MATERIAL CAPSULES: MATERIAL	TYPE OF SAMPLES: BURST TEST	TYPE OF SAMPLES: TENSILE	TYPE OF SAMPLES: CREEP RUPTURE
X011 (CONT'D)		GE					UO_2-20PuO_2	9- F-4-A F-4-D F-4-E F-4-F F-4-G F-4-H F-4-J F-4-K F-4-L	16.9	15.5	6.11	5.81	2739	1.67					
		PNWL					PuO_2-S/S	2- 5P-9 5P-12	10.9	7.1	7.10	7.04	2739	1.94					
		PNWL					UO_2-S/S	1- 5U-14	5.6	5.6	5.78		2739	1.58					
X012	4B2	NuMEC	20,600		8-10-66		UO_2-20PuO_2	19- C-1 C-2 C-3 C-4 C-6 C-7 C-8 C-9 C-10 C-11 C-12 C-13 C-14 C-15 C-16 C-17 C-18 C-19 D-5	14.6	12.7	5.74	5.08	610	.35					
X013	3C1	ANL	1,200	1,309	7-17-66	9-7-66									19- As-34 A -35	HAST-X INCO-625 V-20Ti	X X 	X X	

TABLE XIII (CONT'D)

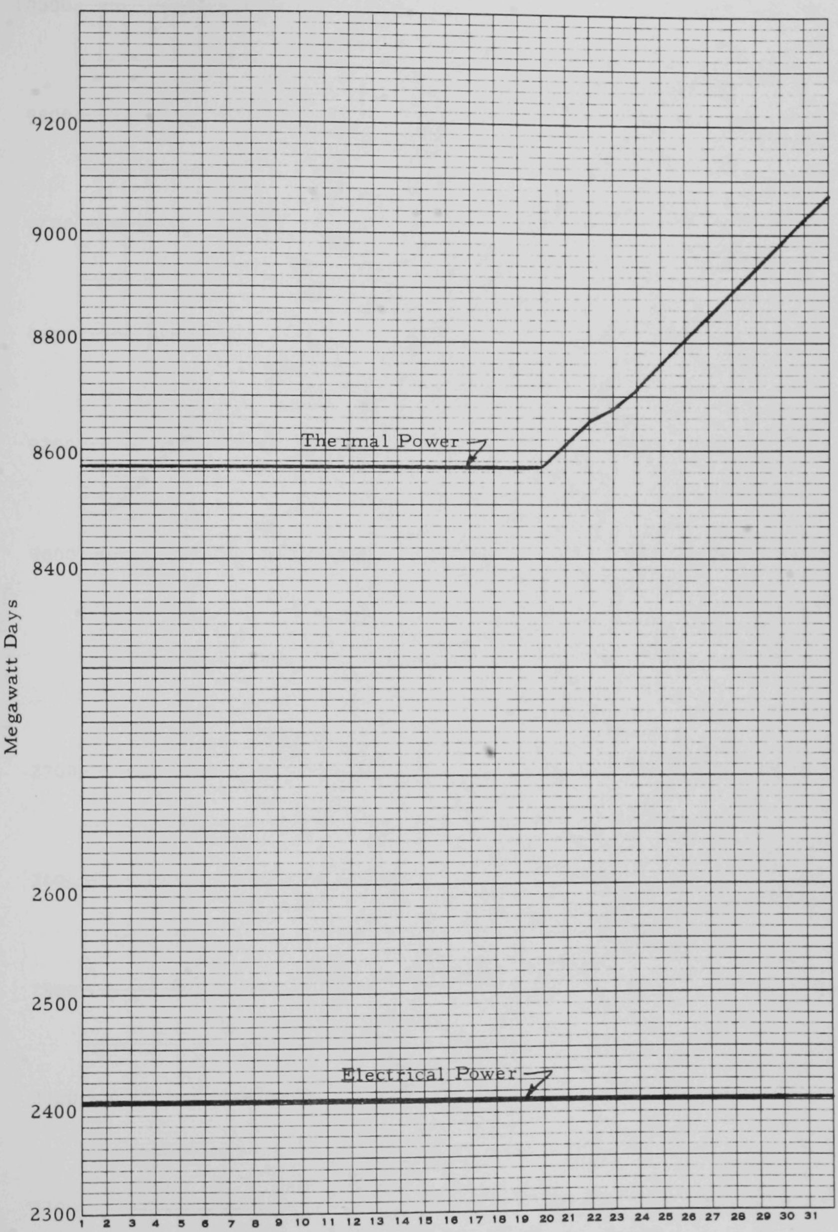
SUBAS-SEMBLY	GRID LOCA-TION	EXPERI-MENTER(S)	GOAL EXPOSURE MWd	ACTUAL FINAL EXPOSURE MWd	DATE INSTALLED	DATE REMOVED	FUEL CAPSULES: FUEL	FUEL CAPSULES: NO. OF CAPS & DESIG-NATION	FUEL CAPSULES: POWER GENERATION kW/FT. MAX.	FUEL CAPSULES: POWER GENERATION kW/FT. MIN.	FUEL CAPSULES: MID-PLANE BURNUP RATES ($^a/o$/Mwd)x10^4 MAX.	FUEL CAPSULES: MID-PLANE BURNUP RATES ($^a/o$/Mwd)x10^4 MIN.	FUEL CAPSULES: STATUS AS OF (9-30-66) MWd	FUEL CAPSULES: STATUS AS OF (9-30-66) %BU(MAX)	MATERIAL CAPSULES: NO. OF CAPS & DESIG-NATION	MATERIAL CAPSULES: MATERIAL	TYPE OF SAMPLES: BURST TEST	TYPE OF SAMPLES: TENSILE	TYPE OF SAMPLES: CREEP RUPTURE
X013 (CONT'D)															As-36	INCO-625	X		
																V-20Ti		X	
															As-37	HAST-X	X	X	
															As-38	V-15Ti-7.5 CR	X		
																V-20Ti		X	
															As-39	V-15Ti-7.5 CR	X		
																V-20Ti		X	
															As-40	V-15Ti-7.5 CR	X		
																V-20Ti		X	
															As-41	V-15Ti-7.5 CR	X		
																V-20Ti		X	
															As-42	V-15Ti 7.5 CR	X	X	
															As-43	V-15Ti 7.5 CR	X	X	
															As-44	V-15Ti 7.5 CR	X	X	
															AS-45	INCO-625	X		
																V-15Ti 7.5 CR		X	
															As-46	HAST-X	X	X	
															As-47	304	X	X	
															As-48	304	X	X	
															As-49	304	X	X	
															As-54	V-15Ti 7.5 CR	X	X	
															As-55	INCO-625	X		
																V-15Ti 7.5 CR		X	
		PNWL													BG-1	GRAPHITE			
X014	2D1	PNWL	**		7-17-66										19-A-9	304		X	
															A-10	348		X	
															A-11	348		X	
															A-12	304		X	
															A-13	304 & 348		X	

TABLE XIII (CONT'D)

SUBAS-SEMBLY	GRID LOCA-TION	EXPERI-MENTER(S)	GOAL EXPOSURE MWd	ACTUAL FINAL EXPOSURE MWd	DATE INSTALLED	DATE REMOVED	FUEL CAPSULES									MATERIAL CAPSULES				
							FUEL	NO. OF CAPS & DESIG-NATION	POWER GENERATION kW/FT.		MID-PLANE BURNUP RATES $(^a/o/Mwd)\times10^4$		STATUS AS OF (9-30-66)		NO. OF CAPS & DESIG-NATION	MATERIAL	TYPE OF SAMPLES		
									MAX.	MIN.	MAX.	MIN.	MWd	%BU(MAX)			BURST TEST	TENSILE	CREEP RUPTURE
X014 (CONT'D)		GE													L4A	I-800	X	X	
															L4B	I-800	X	X	
															L4E	347	X	X	
															L4F	304	X	X	
															L4G	321	X	X	
		NRL													NRL-1	I-800		X	
																HAST-X		X	
																304, 316		X	
															NRL-2	I-800		X	
																316		X	
															NRL-3	I-800,			
																304		X	
																316		X	
															NRL-4	I-800,			
																304		X	
																316		X	
															NRL-5	INCO-625		X	
																HAST-X,			
																316		X	
		PNWL													BG-2	GRAPHITE			
															BG-3	GRAPHITE			
		GE													MT-5	I-800	X	X	
															MT-6	I-800	X	X	

* REMOVED AT REQUEST OF EXPERIMENTER
** MAXIMUM ATTAINABLE BEFORE CORE HAS REACHED TERMINAL SIZE

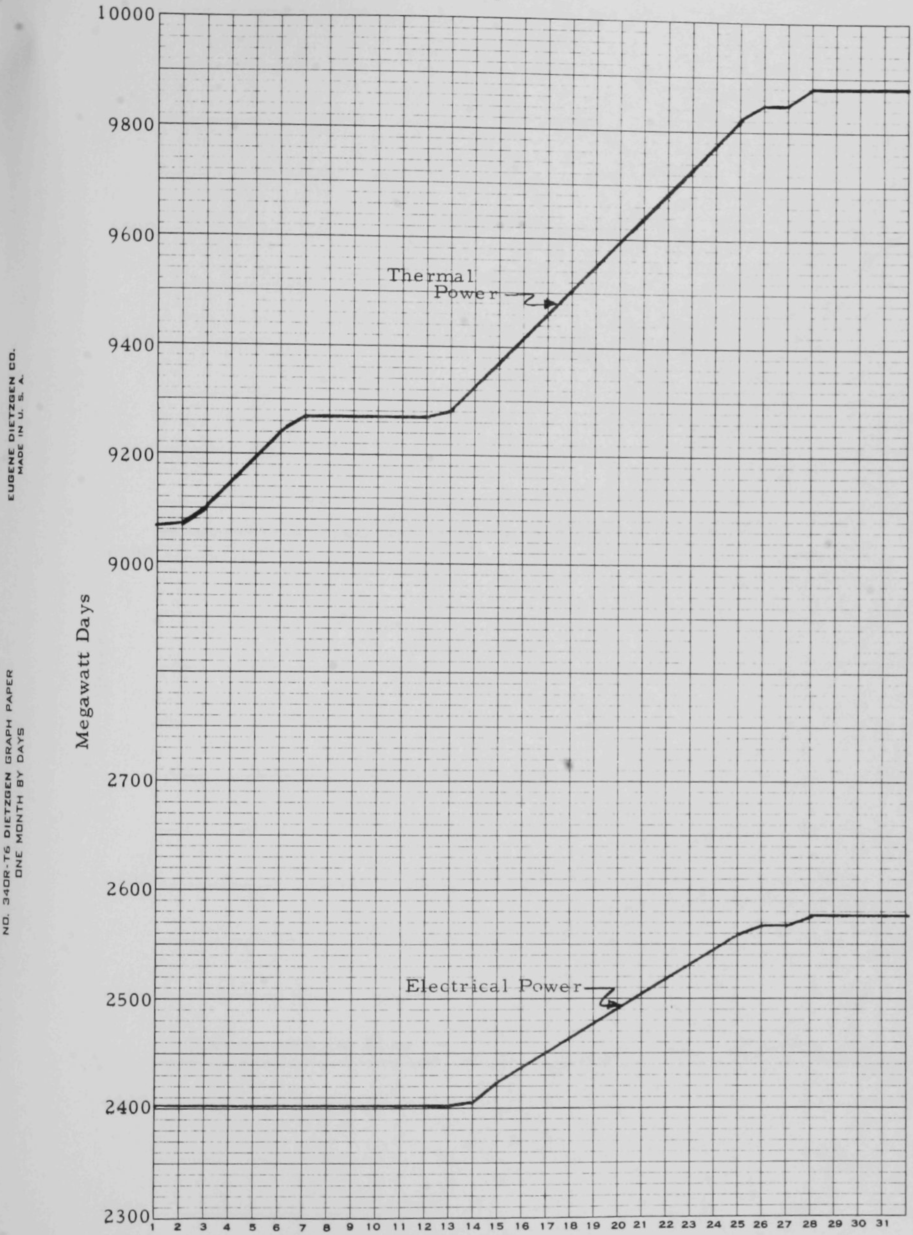
Fig. 1



Integrated Thermal and Electrical Power

MONTH July 1966

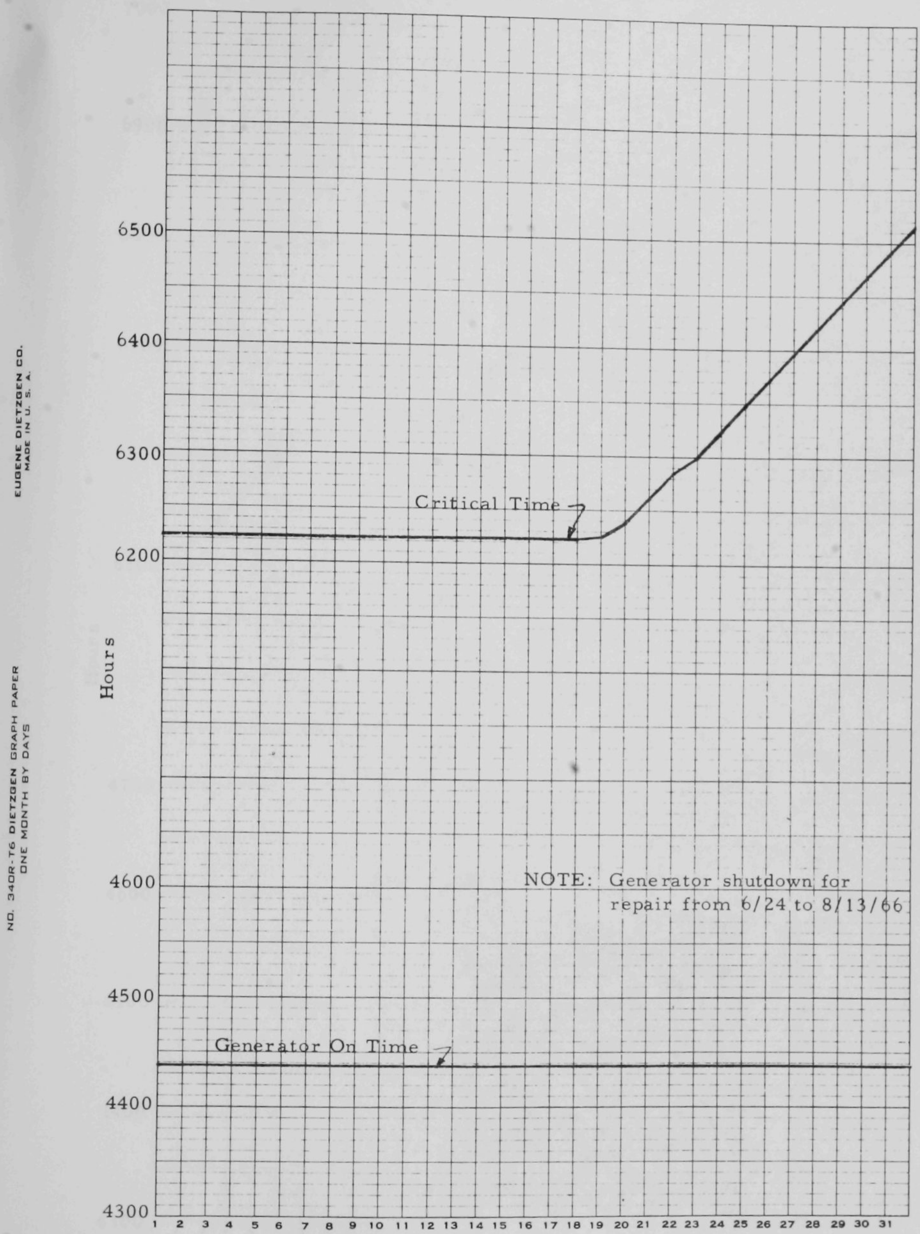
Fig. 2



Integrated Thermal and Electrical Power

MONTH August 19 66

Fig. 3



Critical Time, Generator On Time and Electrical Power Level

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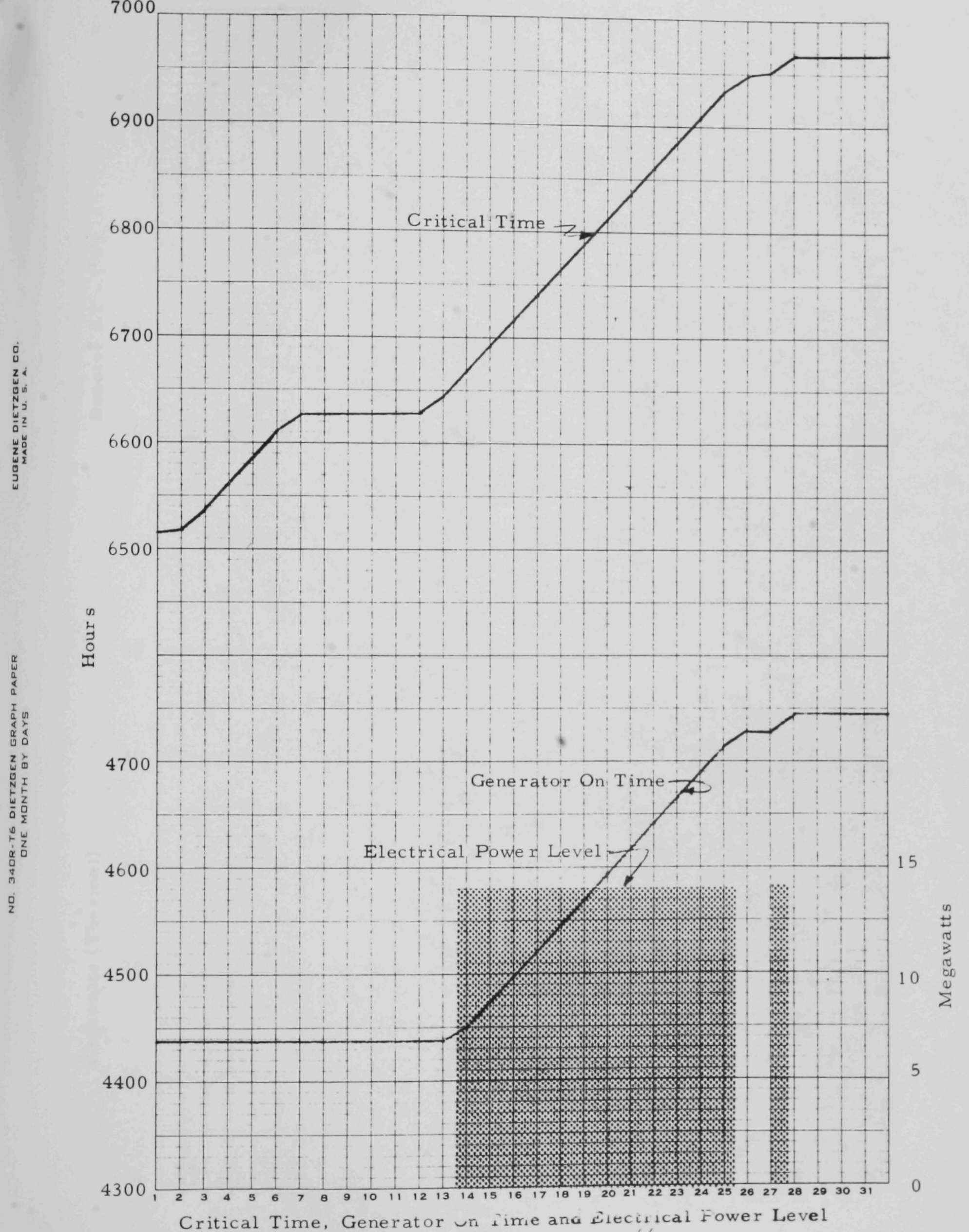
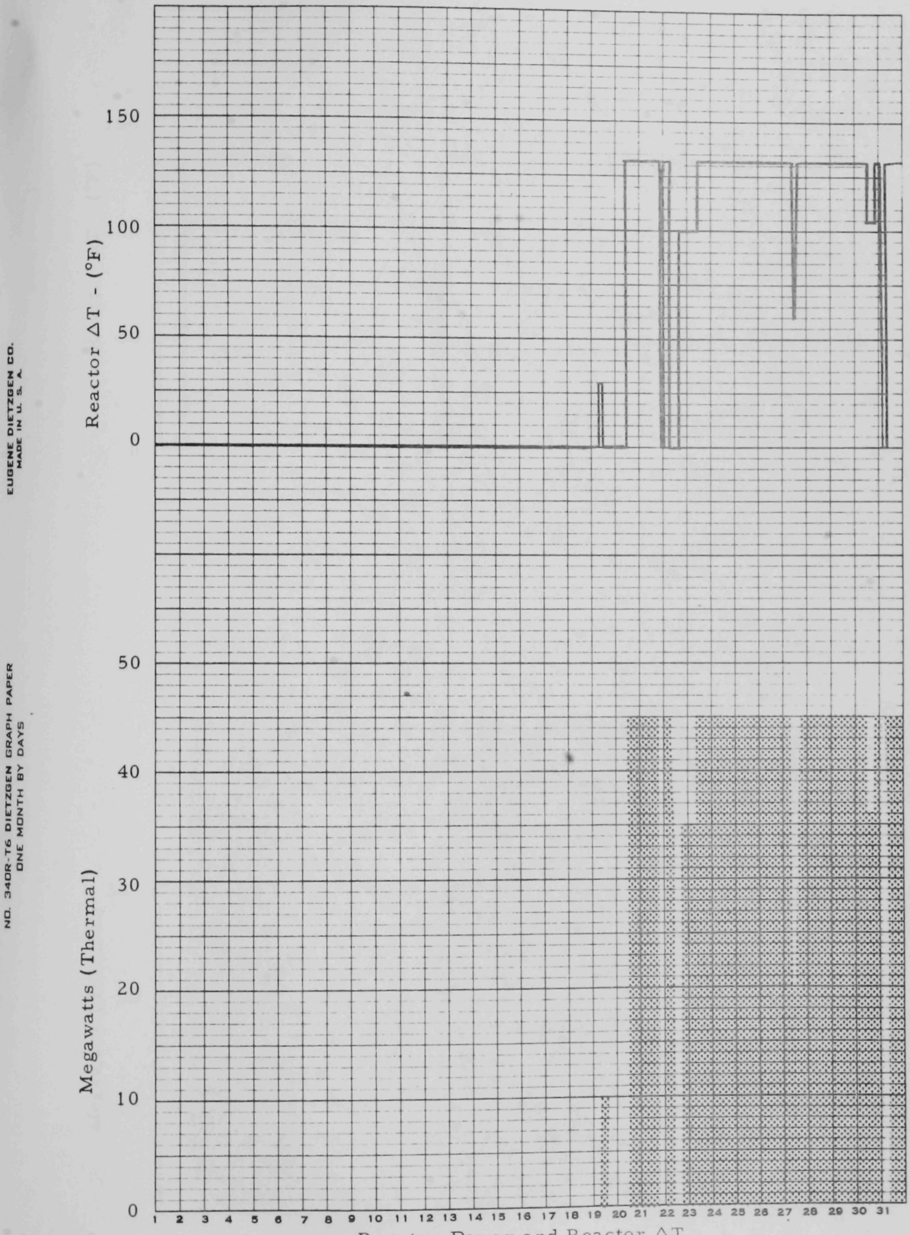
Fig. 4



Critical Time, Generator On Time and Electrical Power Level

MONTH August 1966

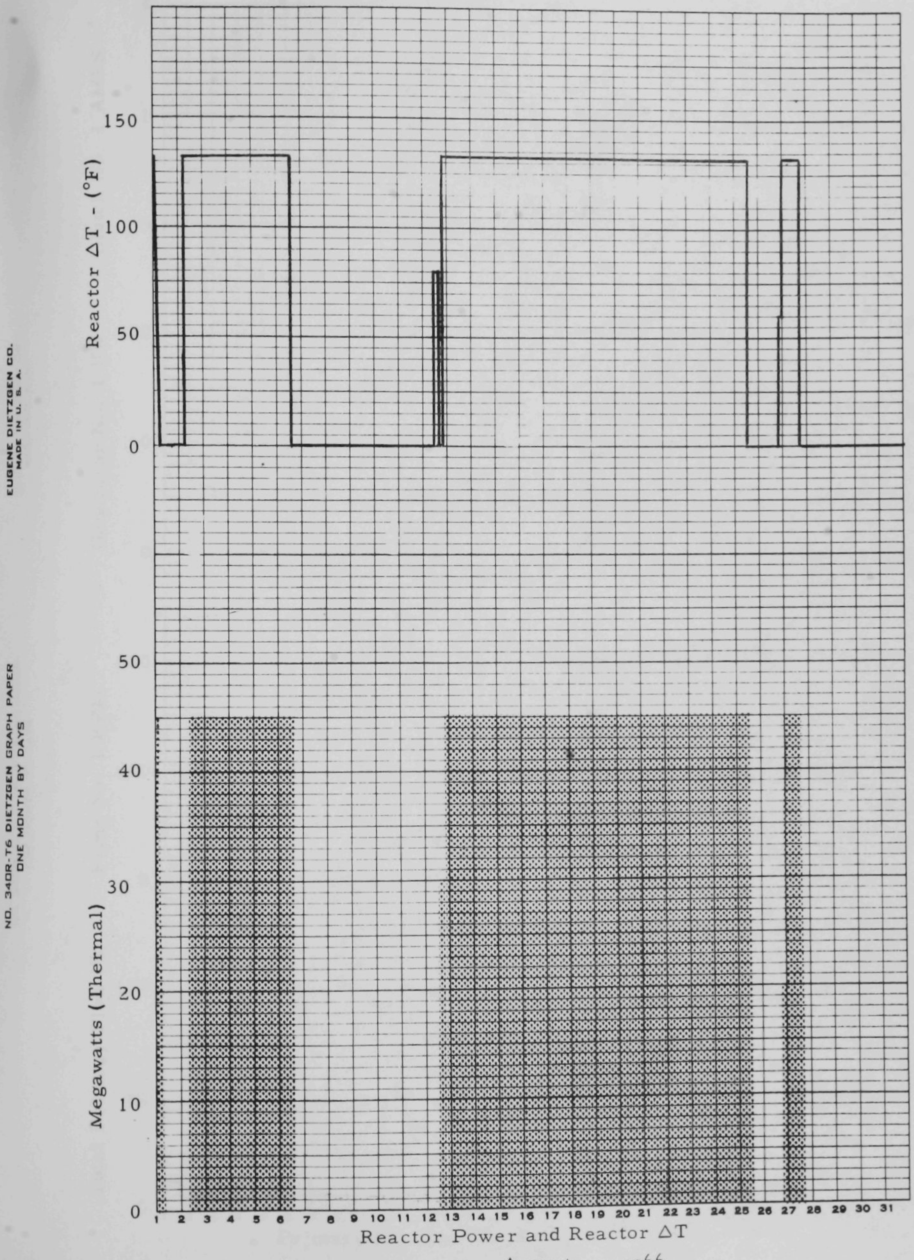
Fig. 5



Reactor Power and Reactor ΔT

MONTH July 1966

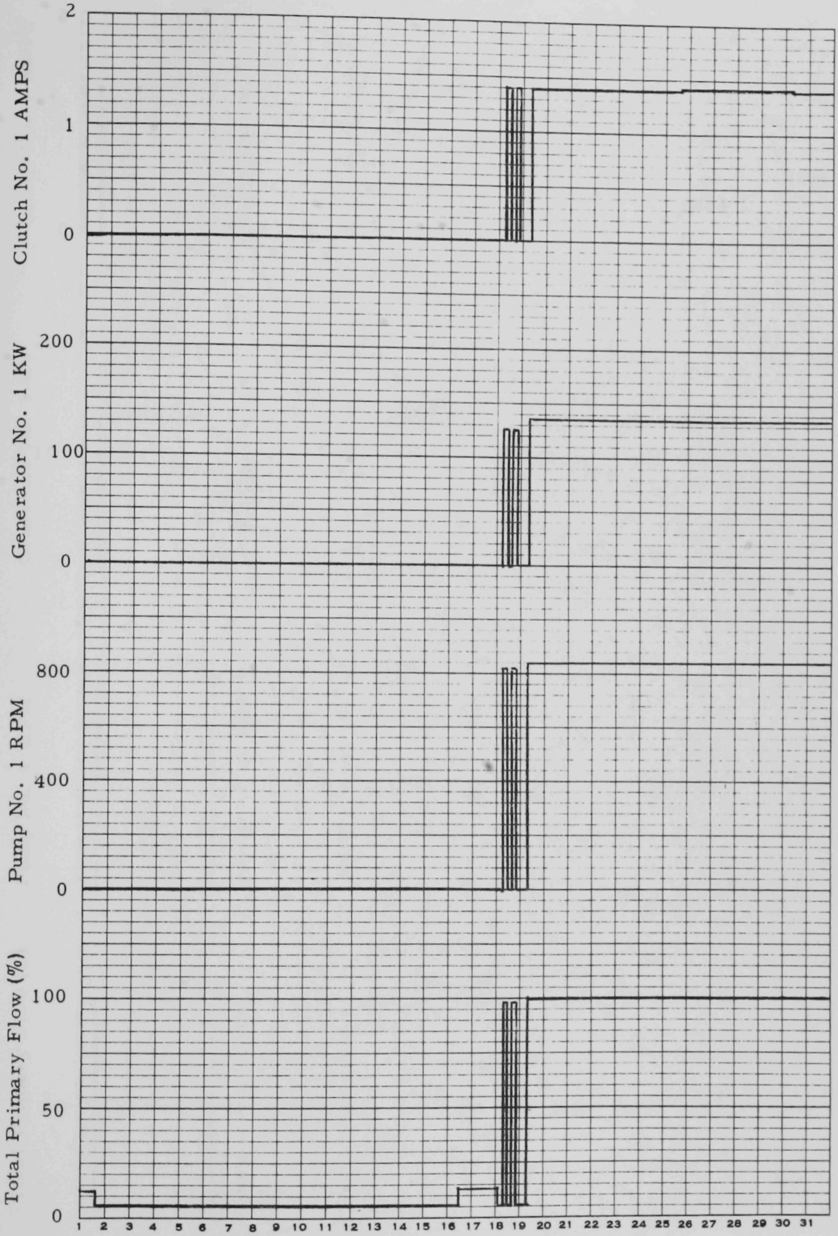
Fig. 6



Reactor Power and Reactor ΔT

MONTH August 1966

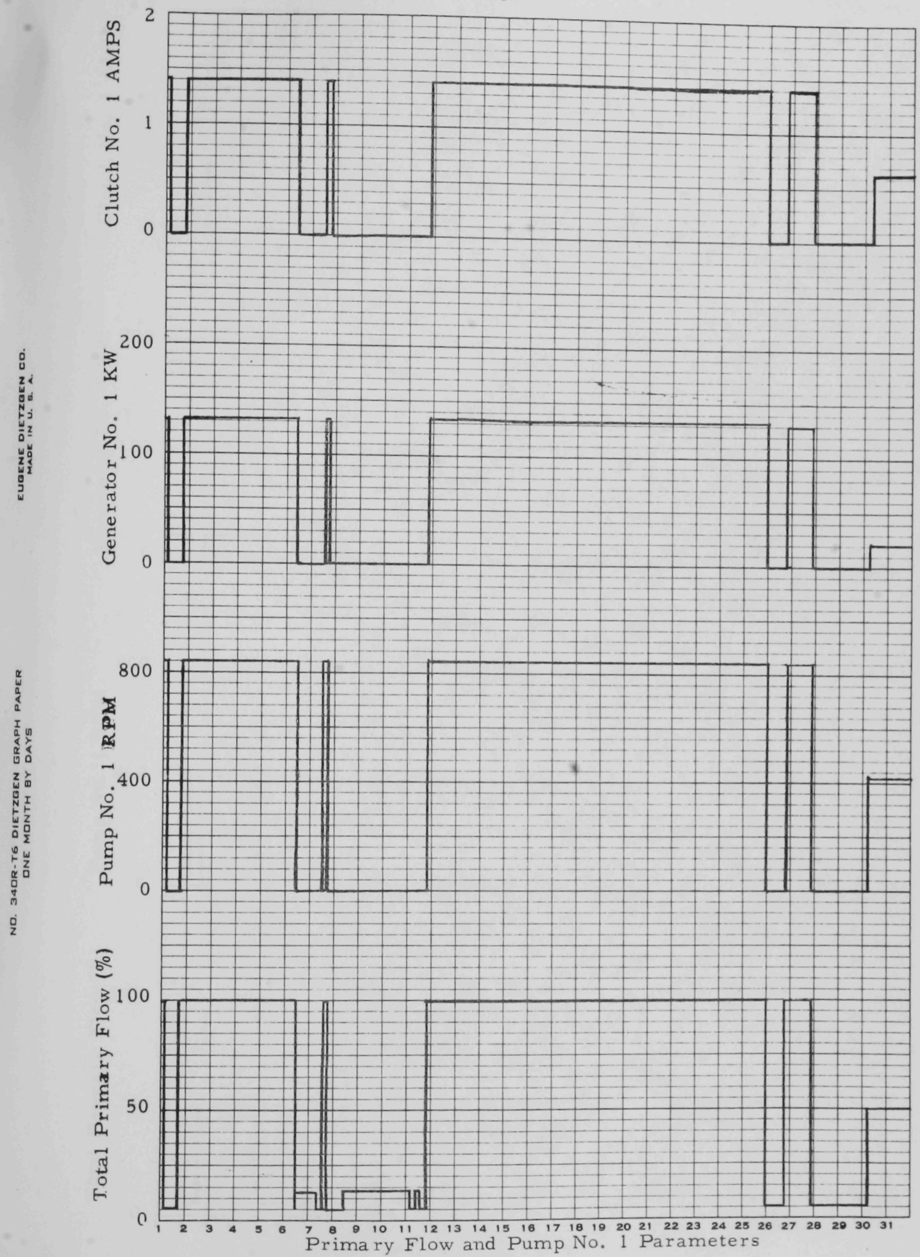
Fig. 7



Primary Flow and Pump No. 1 Parameters

MONTH July 19 1966

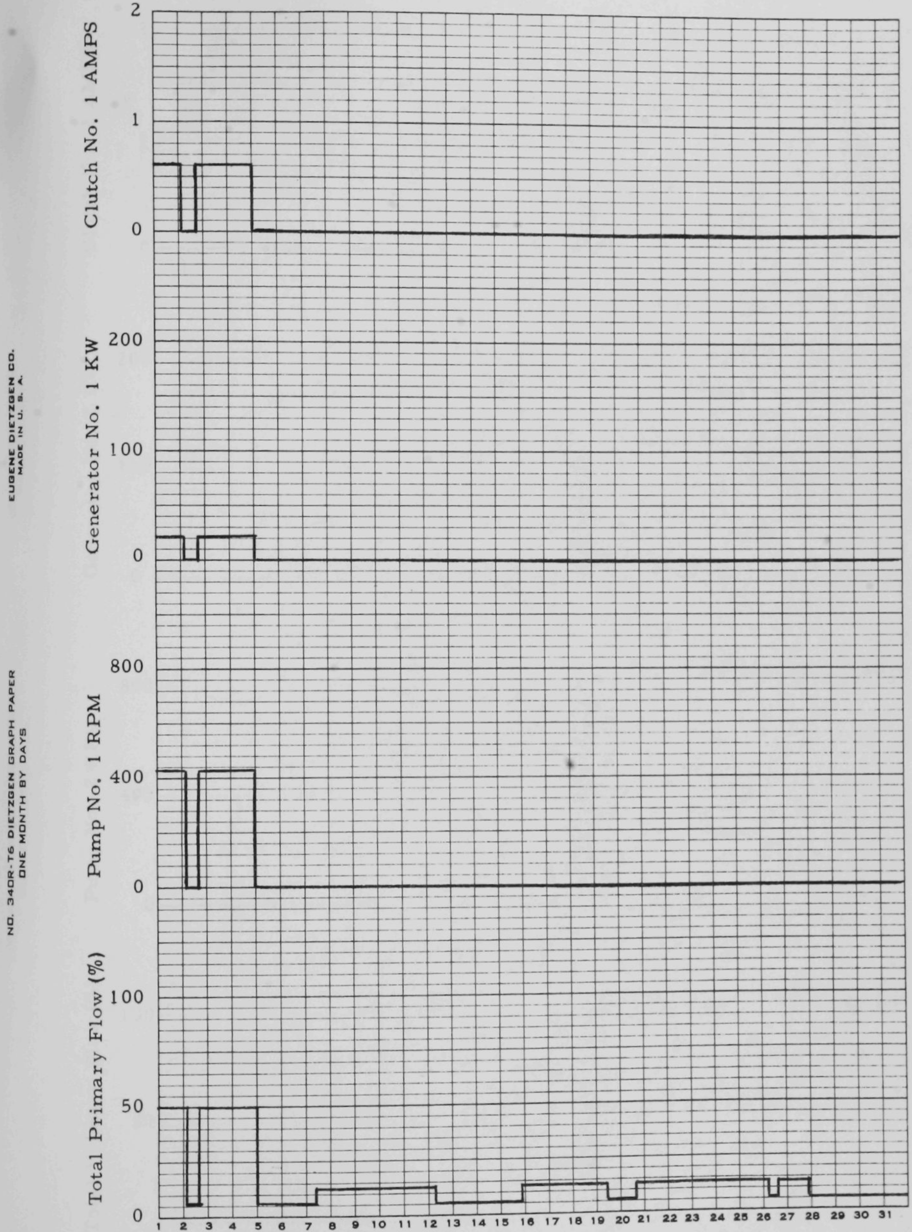
Fig. 8



Primary Flow and Pump No. 1 Parameters

MONTH August 19 66

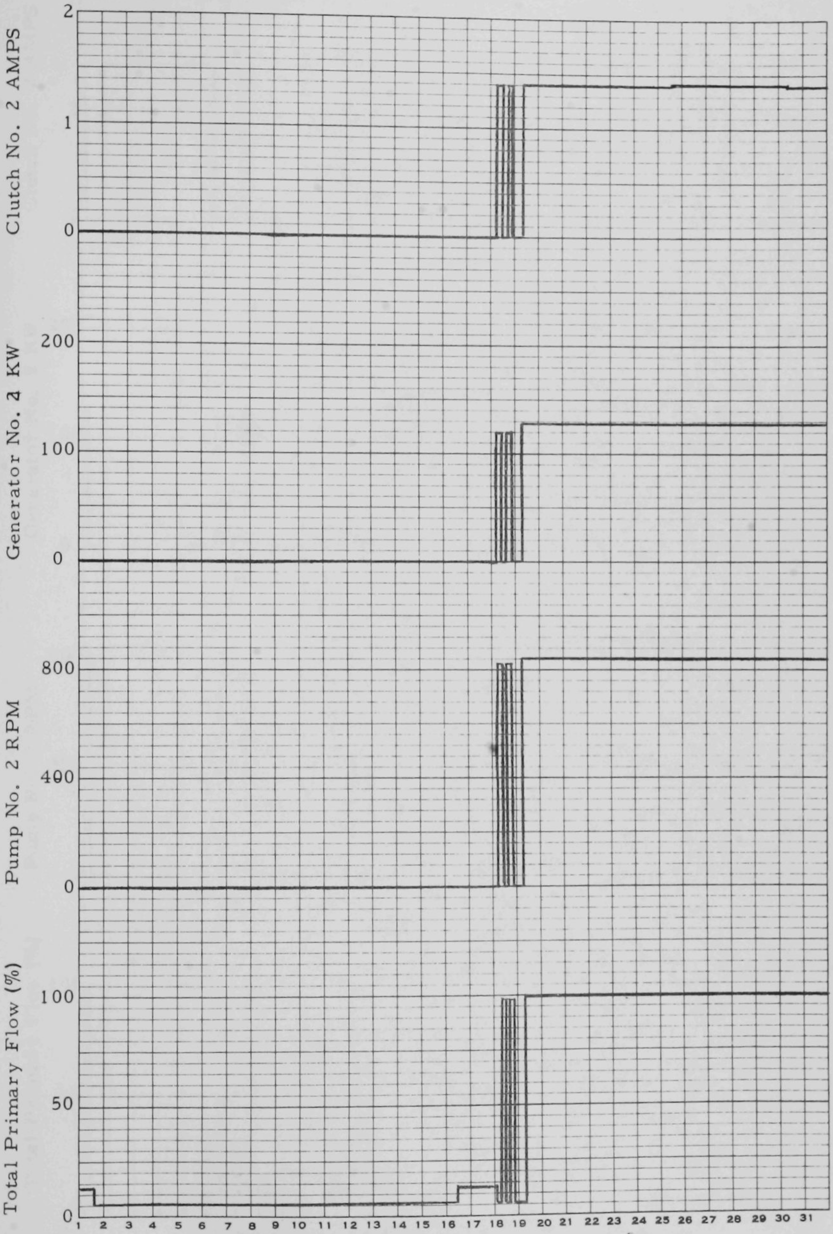
Fig. 9



Primary Flow and Pump No. 1 Parameters

MONTH September 1966

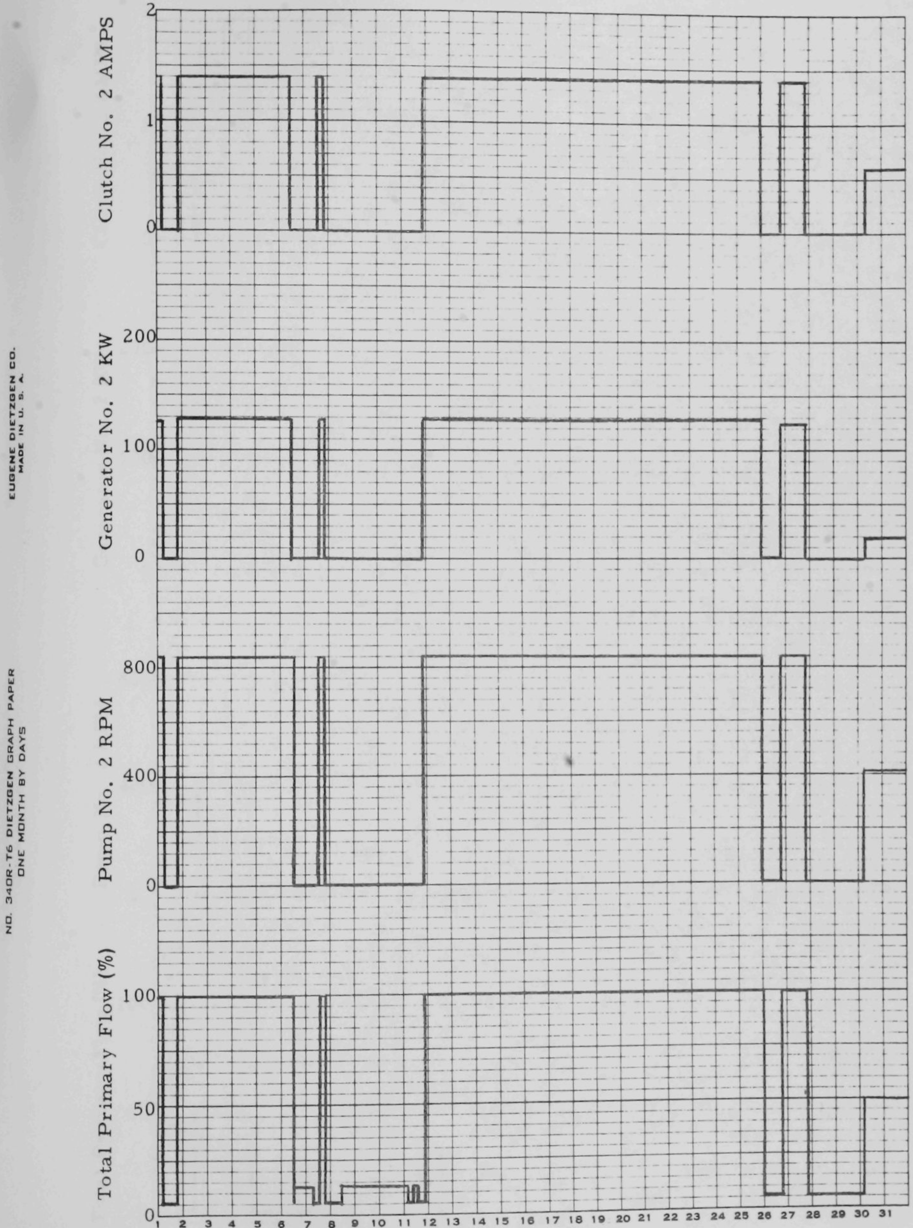
Fig. 10



Primary Flow and Pump No. 2 Parameters

MONTH July 19 66

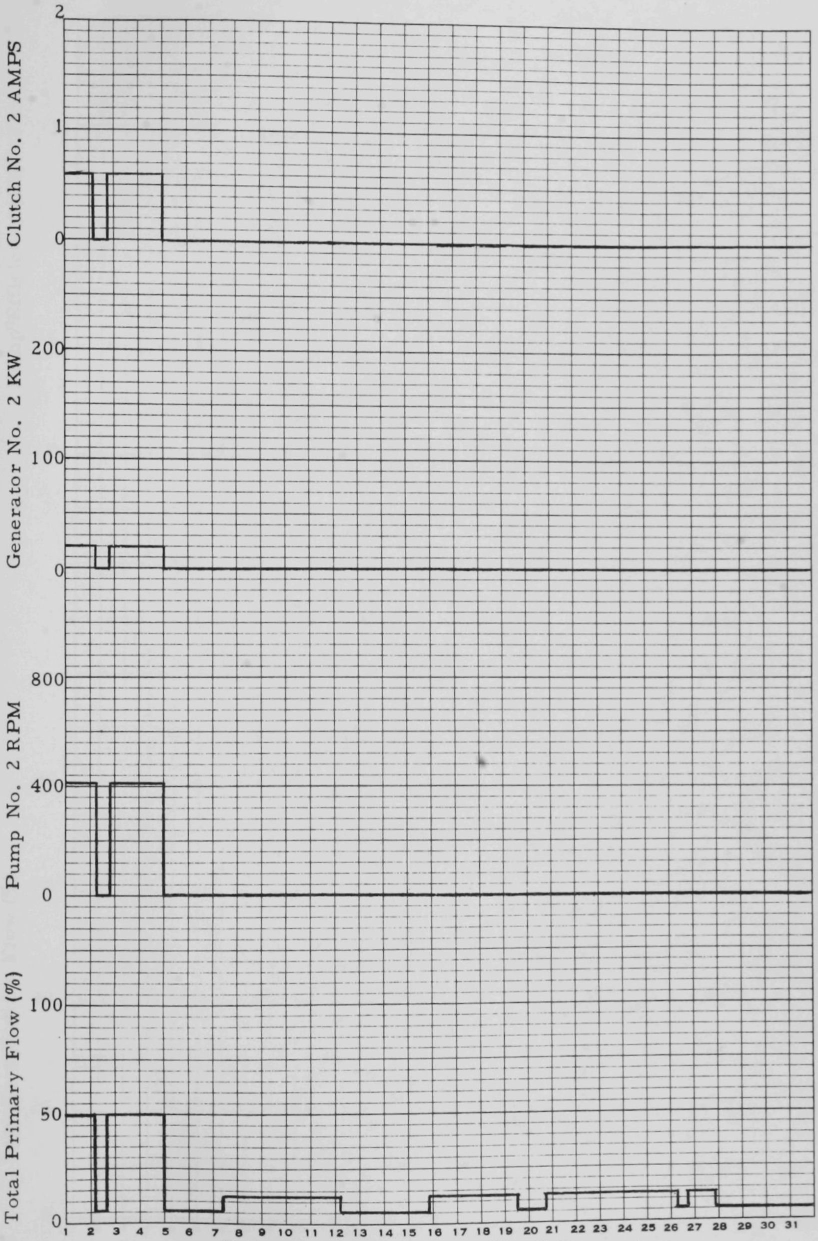
Fig. 11



Primary Flow and Pump No. 2 Parameters

MONTH August 19 66

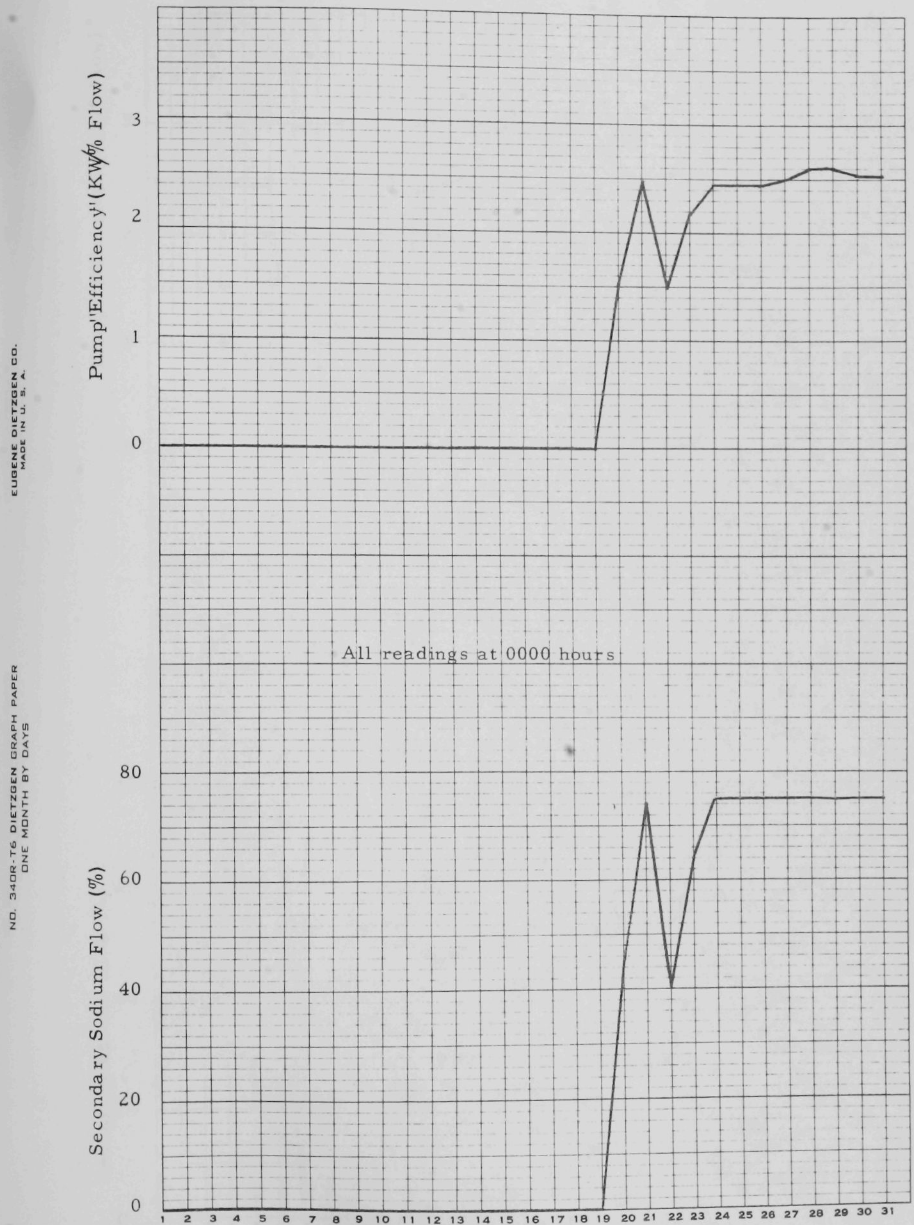
Fig. 12



Primary Flow and Pump No. 2 Parameters

MONTH September 1966

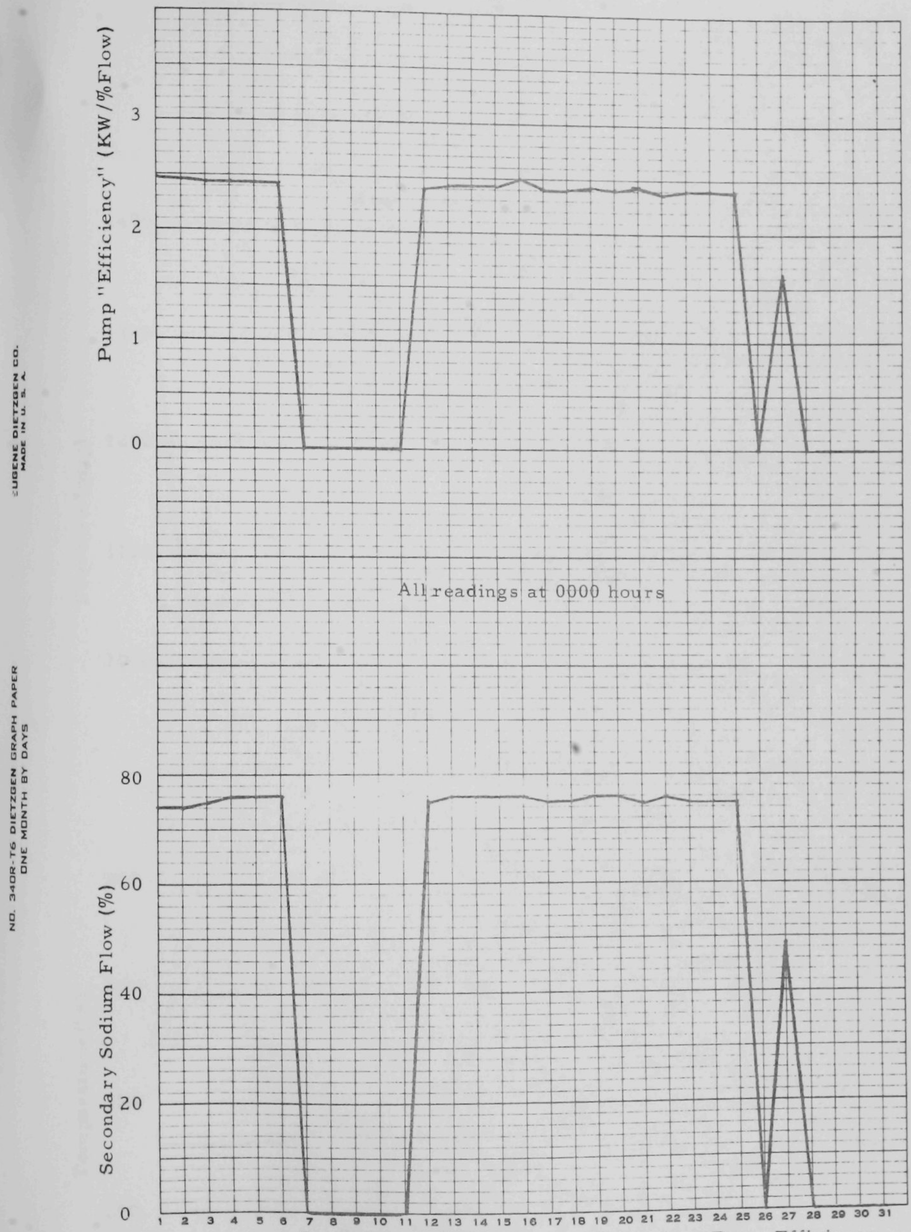
Fig. 13



Secondary Sodium Flow and Secondary Pump Efficiency

MONTH July 19 66

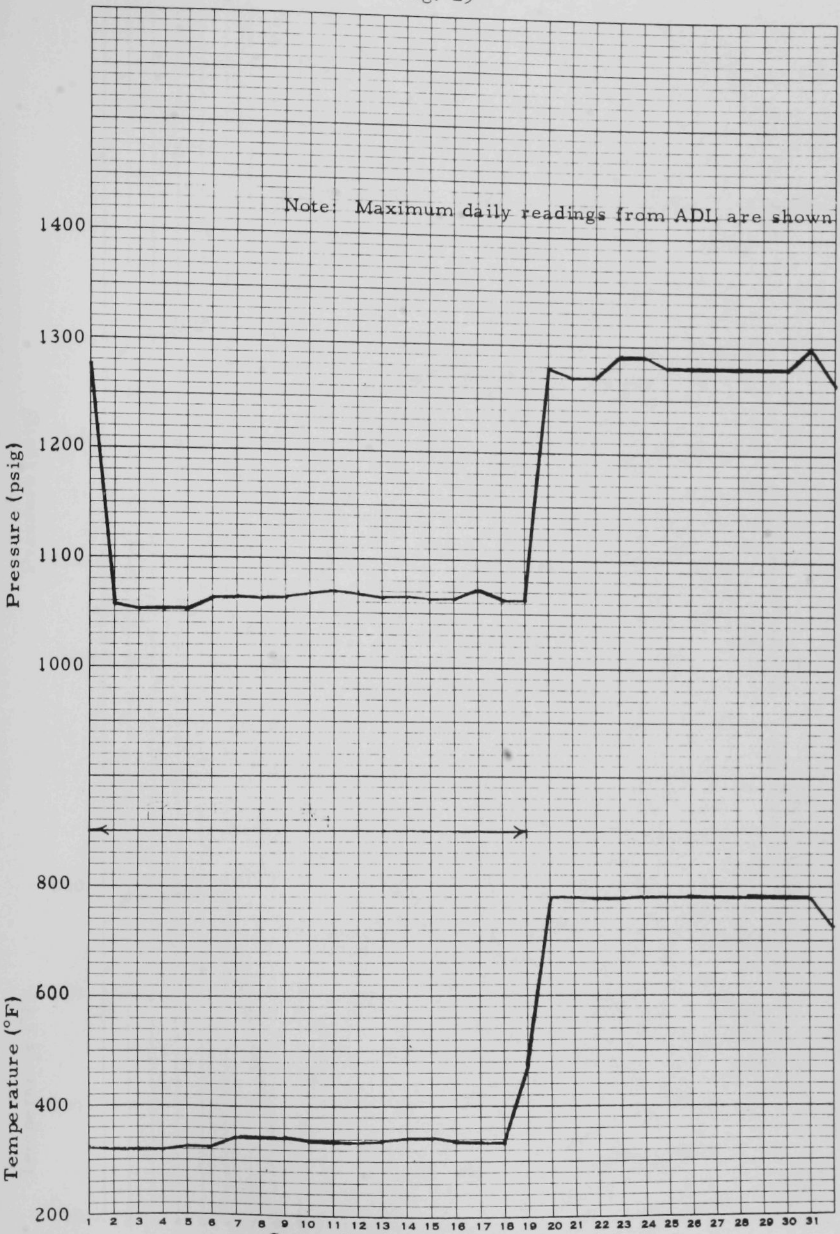
Fig. 14



Secondary Sodium Flow and Secondary Pump Efficiency

MONTH August 19 66

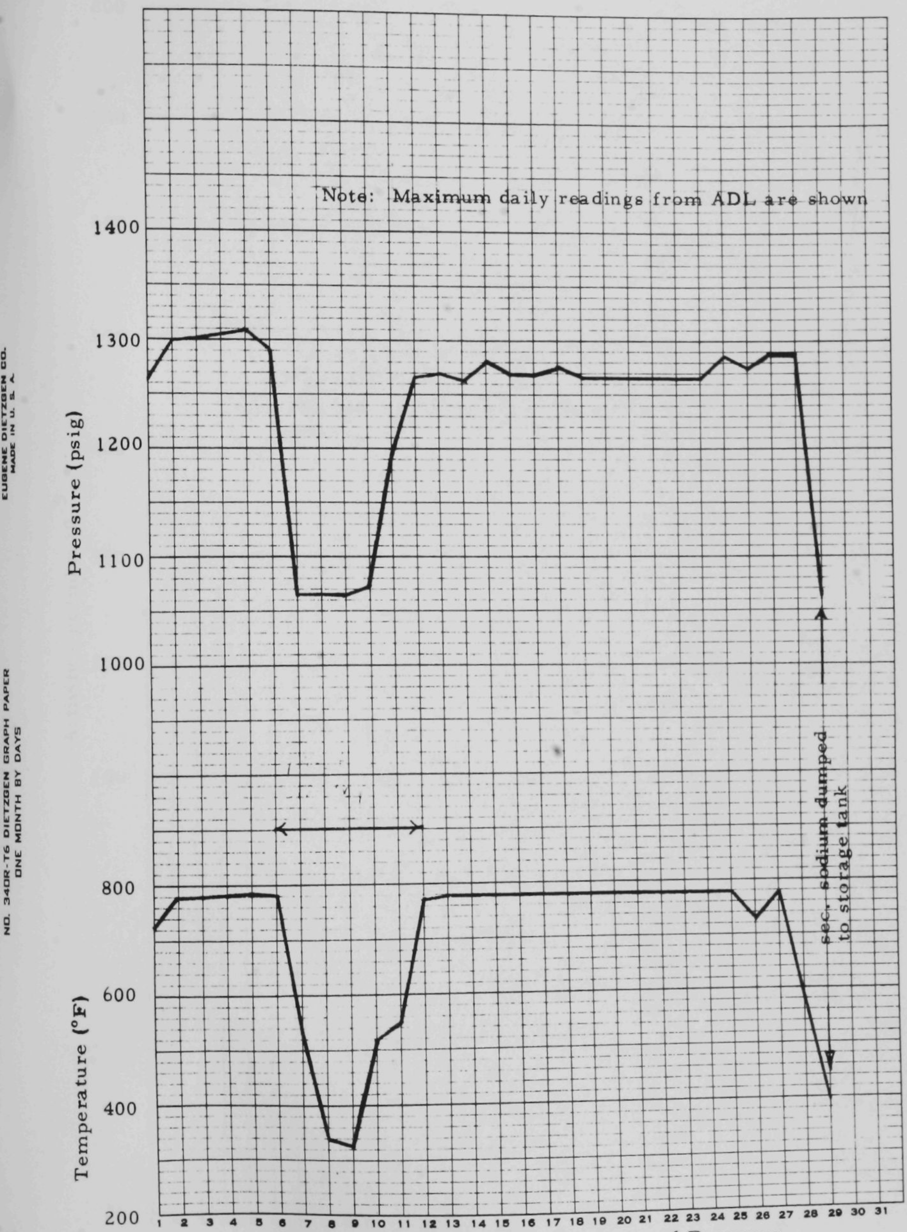
Fig. 15



Steam Header Temperature and Pressure

MONTH July 19 66

Fig. 16



Steam Header Temperature and Pressure

MONTH August 19 66

Fig. 17

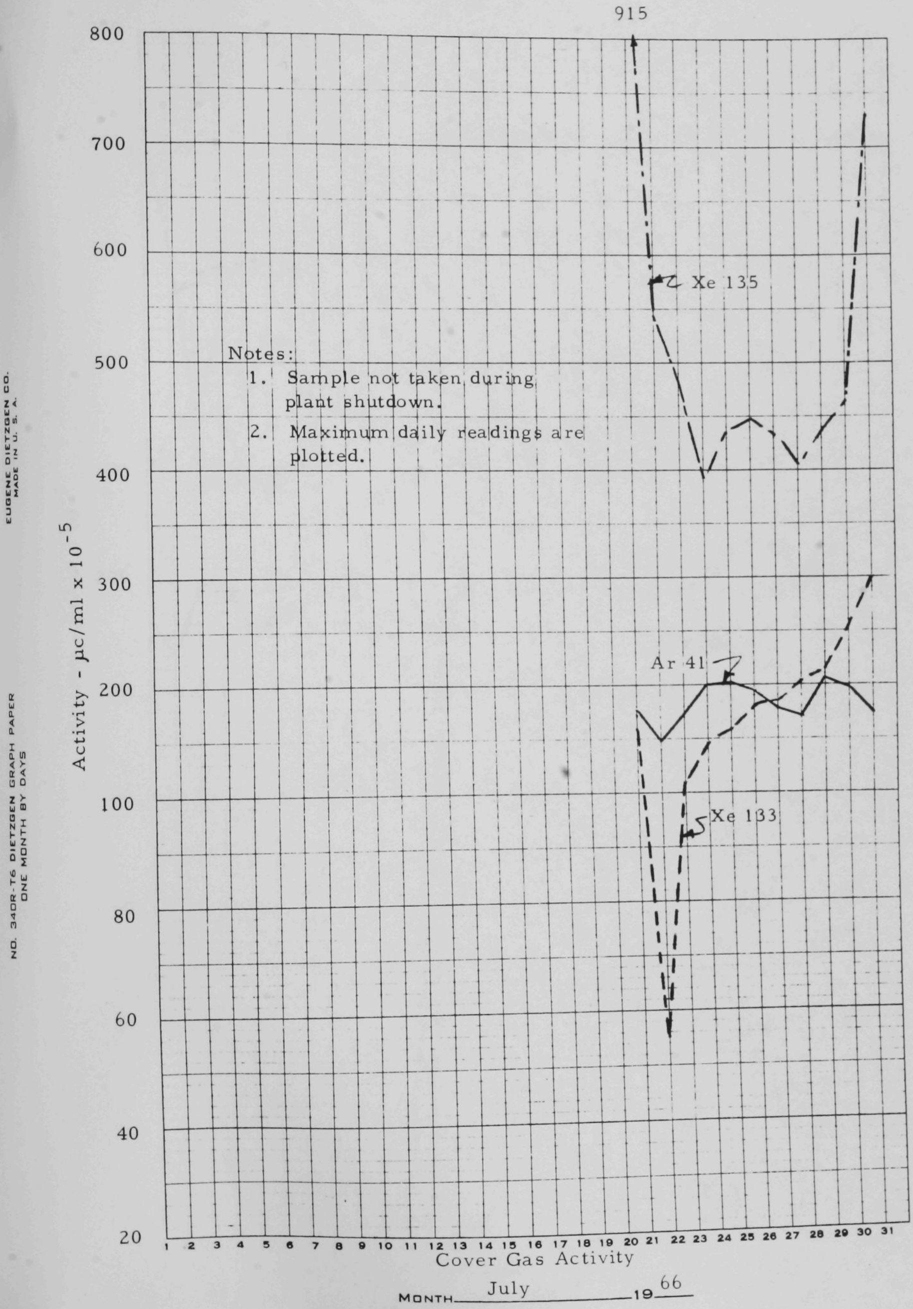
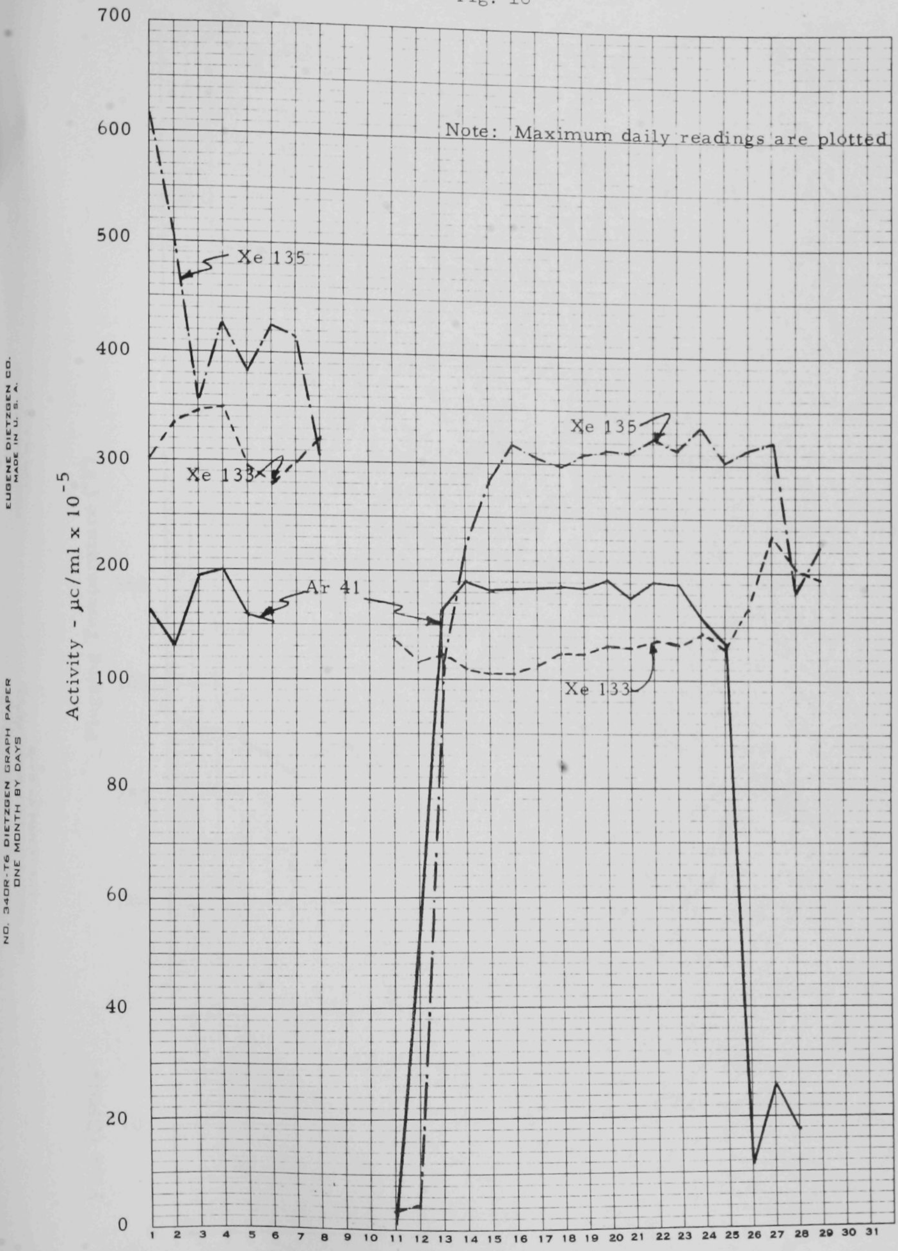

915
800
700
600
500
400
300
200
100
80
60
40
20
Activity - µc/ml x 10^-5
Xe 135
Notes:
1. Sample not taken during plant shutdown.
2. Maximum daily readings are plotted.
Ar 41
Xe 133
1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31
Cover Gas Activity
MONTH July 1966
NO. 340R-T6 DIETZGEN GRAPH PAPER
ONE MONTH BY DAYS
EUGENE DIETZGEN CO.
MADE IN U. S. A.

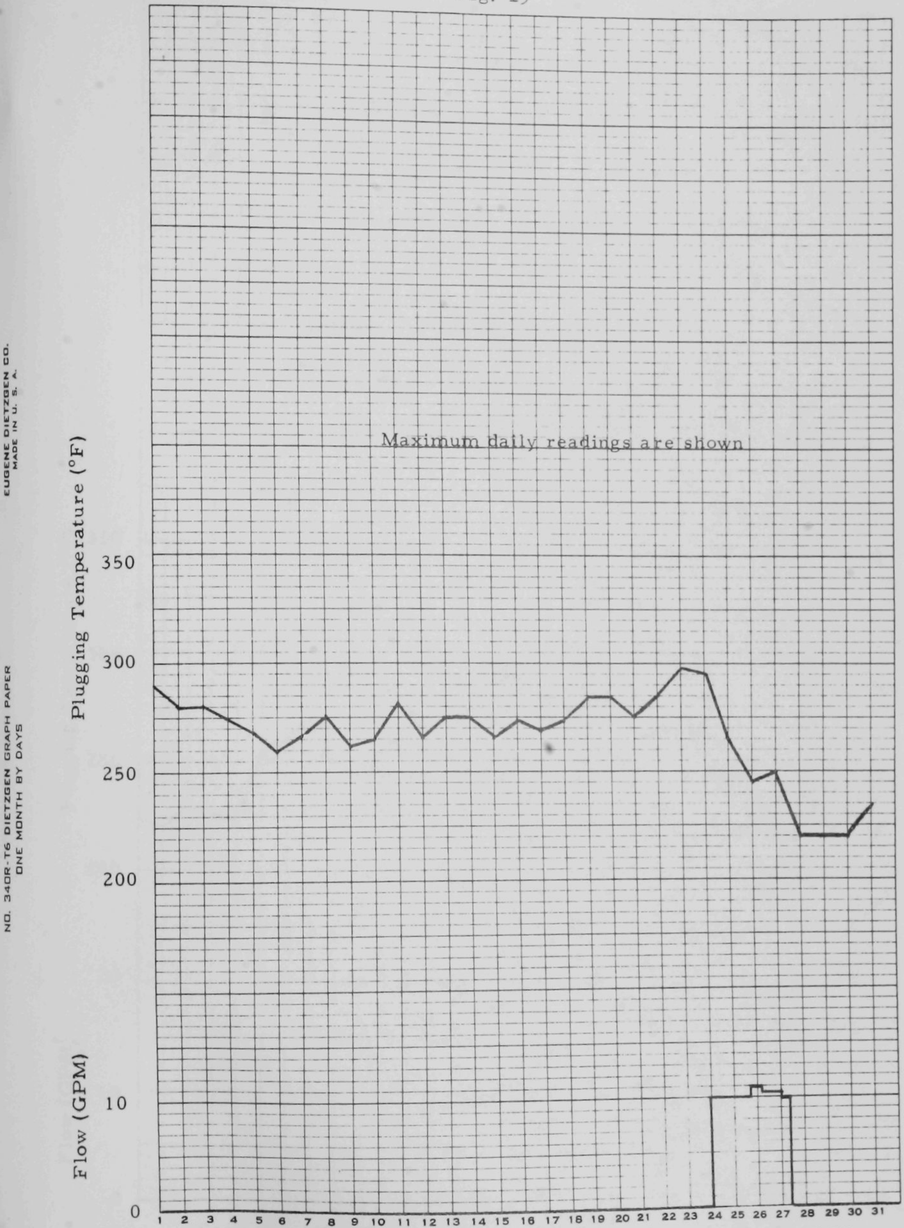
Fig. 18



Cover Gas Activity

MONTH August 1966

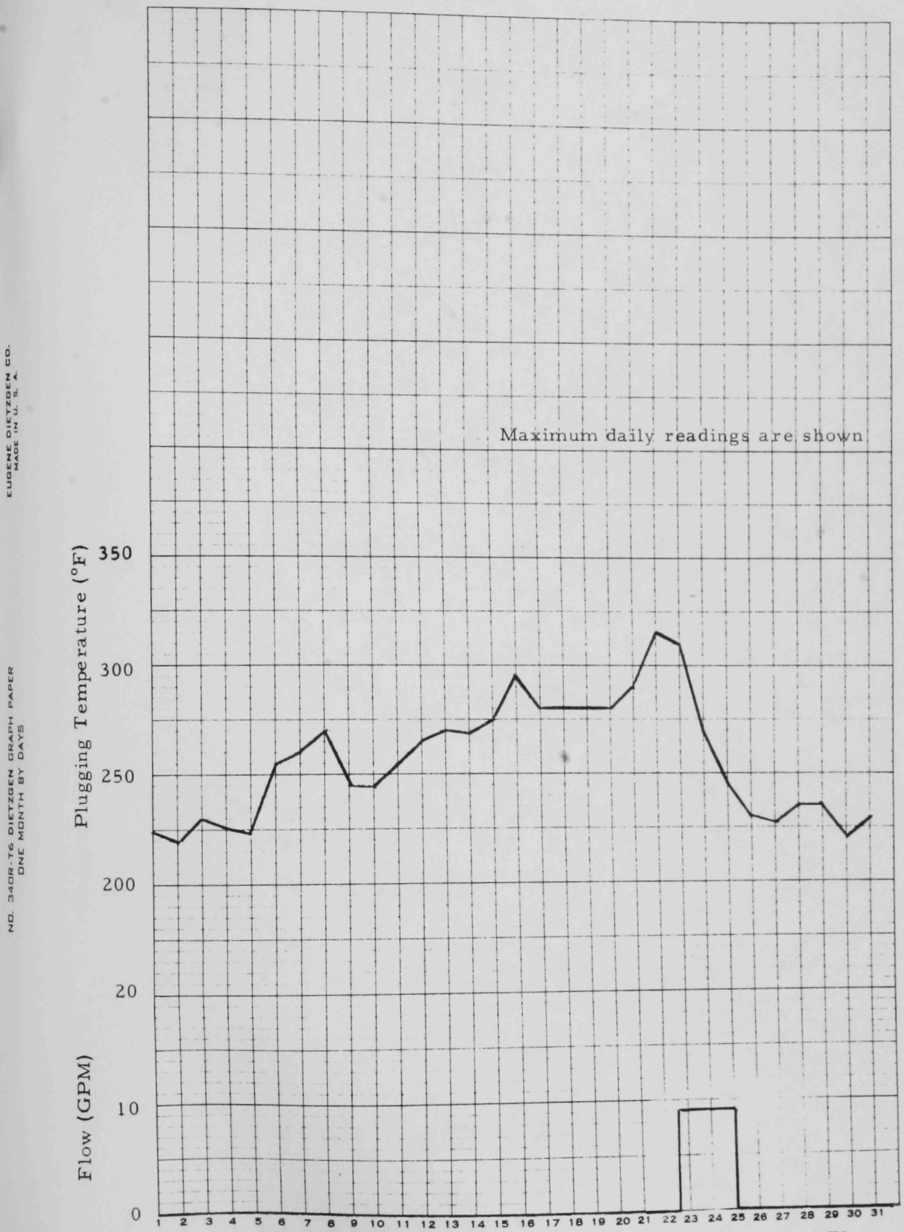
Fig. 19



Secondary Purification System Plugging Temperature and Flow

MONTH July 19 66

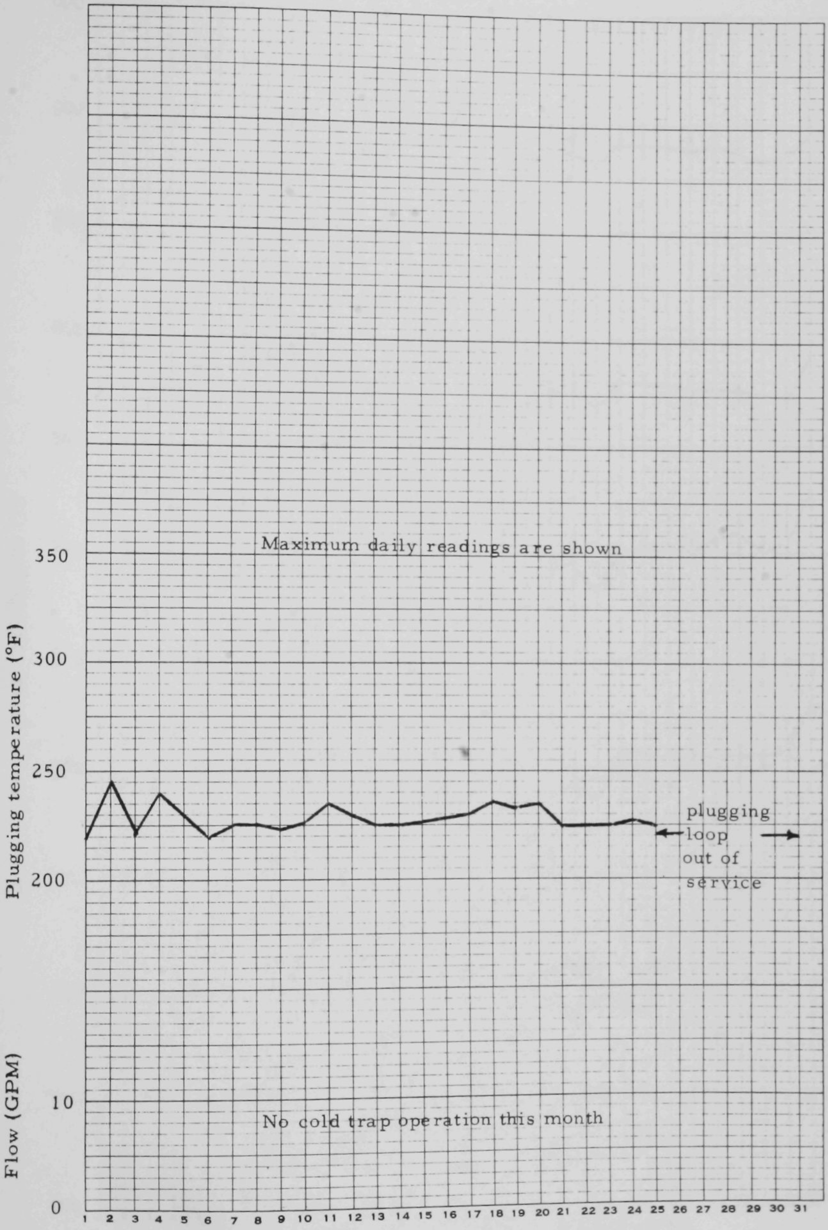
Fig. 20



Secondary Purification System Plugging Temperature and Flow

MONTH August 19 66

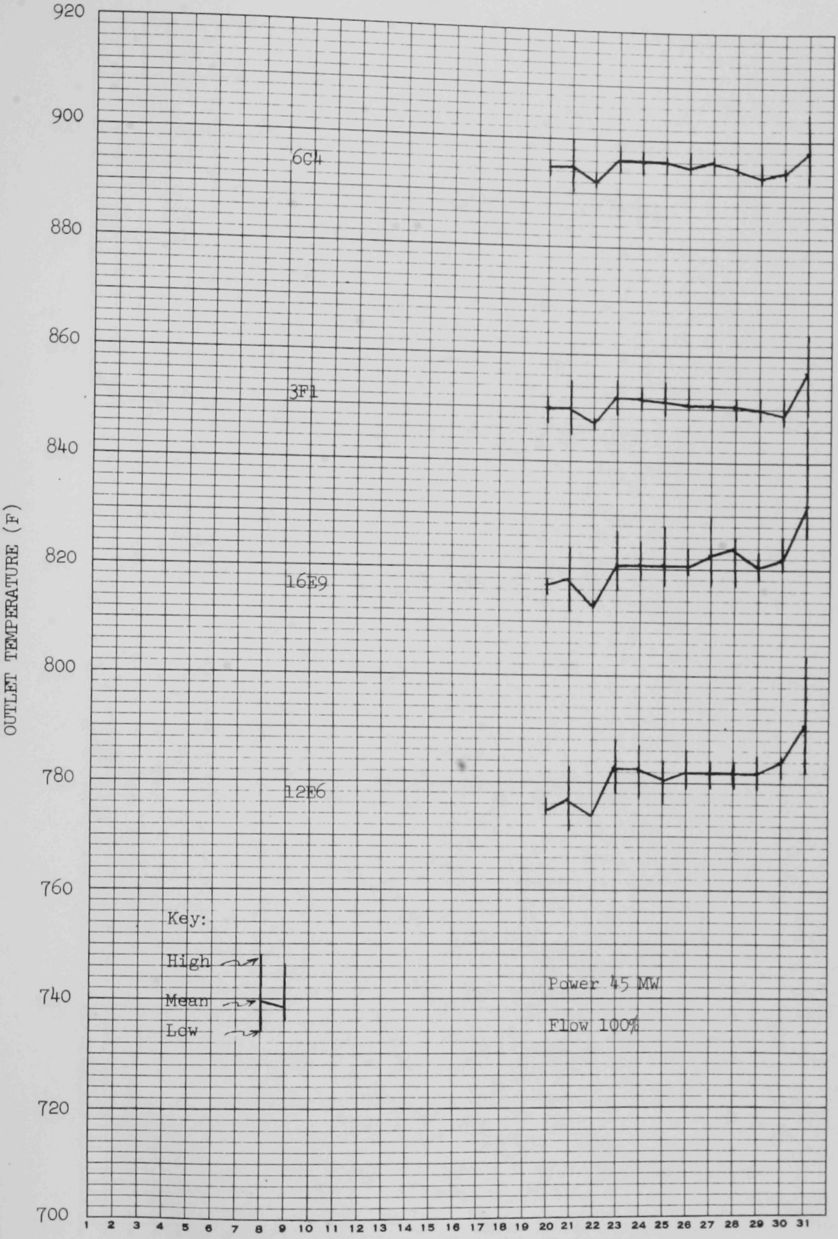
Fig. 21



Secondary Purification System Plugging Temperature and Flow

MONTH September 19 66

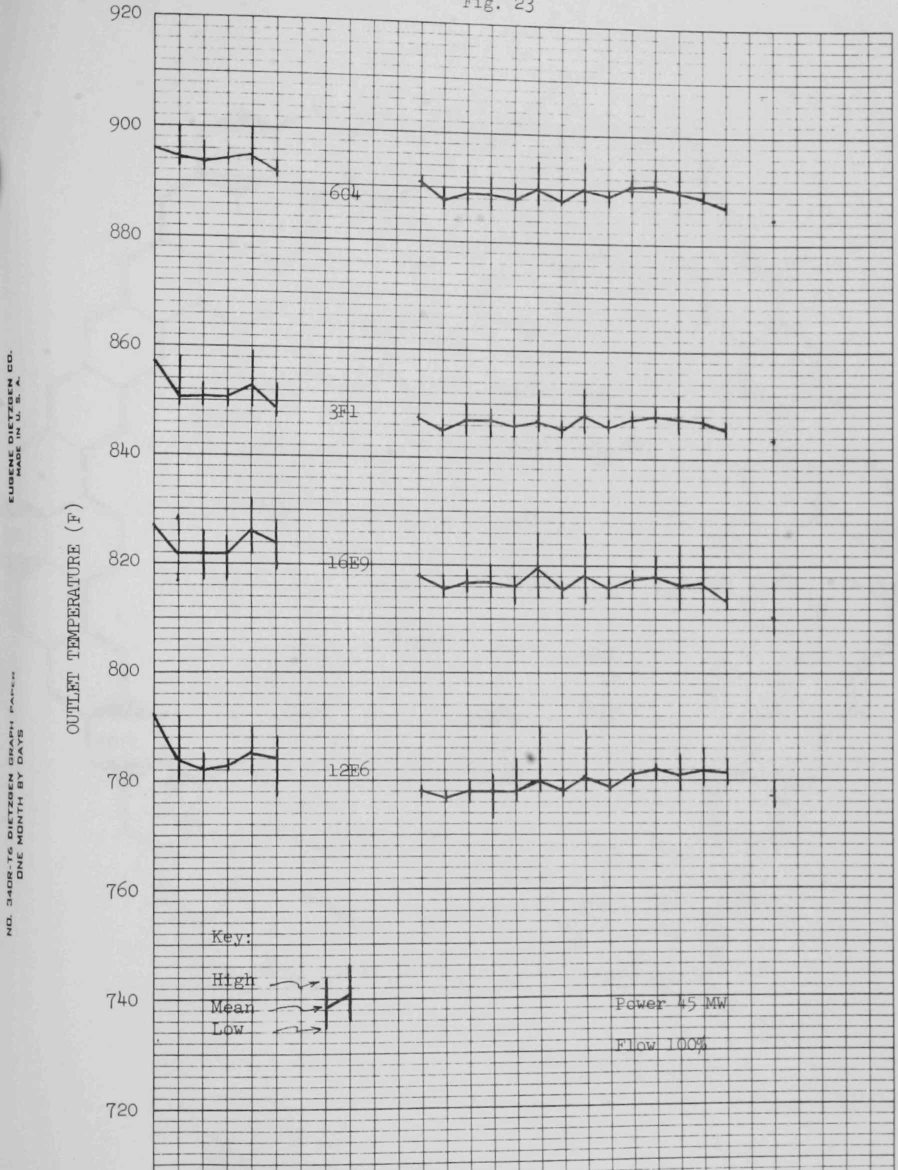
Fig. 22



STEADY STATE SUBASSEMBLY OUTLET TEMPERATURE

MONTH July 1966

Fig. 23



STEADY STATE SUBASSEMBLY OUTLET TEMPERATURE

MONTH August 1966

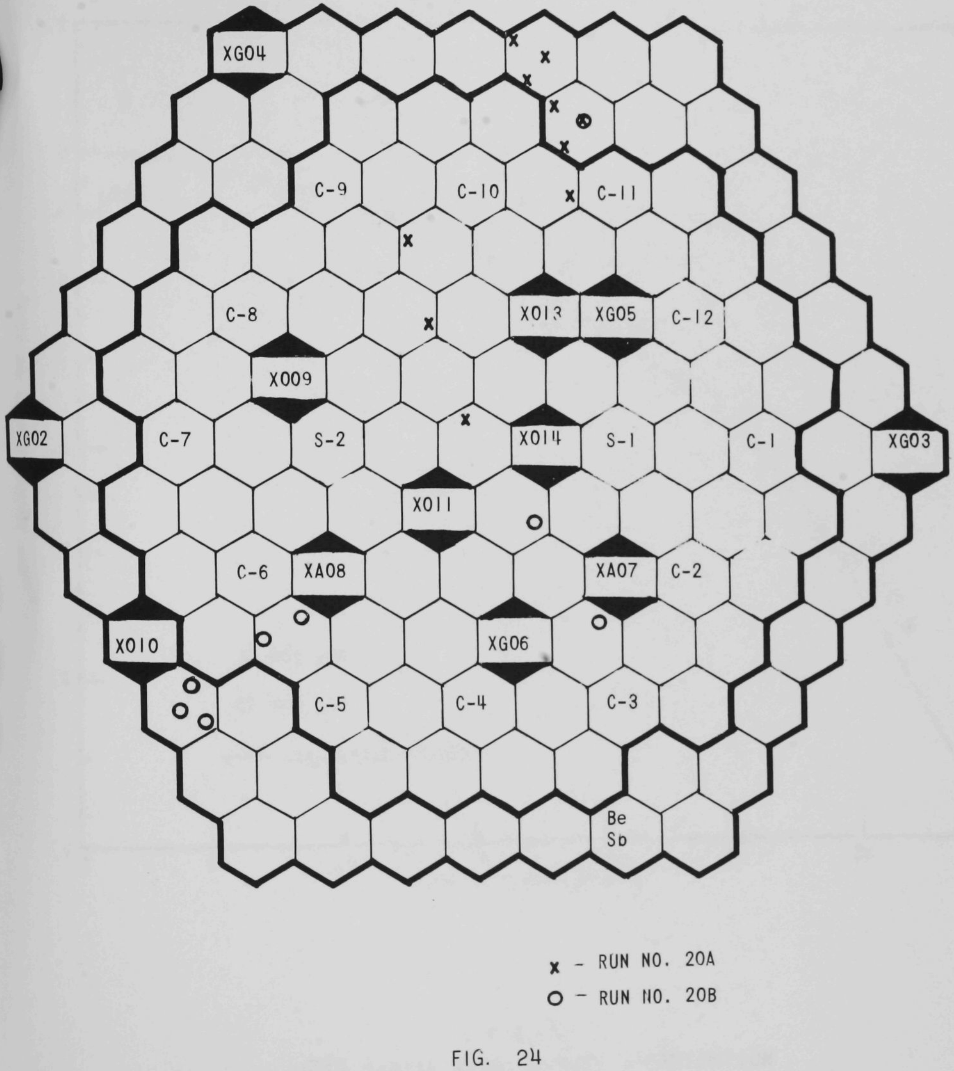


FIG. 24

LOCATION OF WIRE DETECTORS

80-SUBASSEMBLY CORE, EBR-II

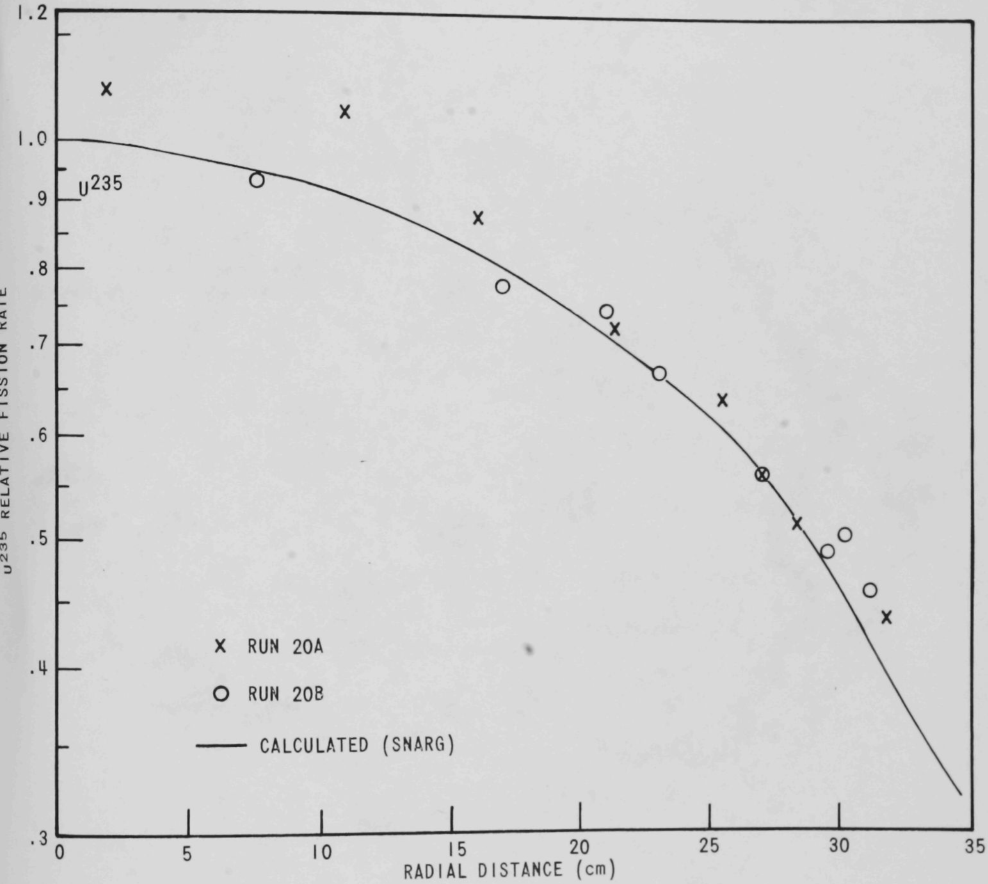


FIG. 25
U^{235} RADIAL FISSION RATE DISTRIBUTION

NOTE: CONTROL ROD #9 (C-9) CONTAINED STAINLESS STEEL PINS ONLY

KEY:
D DRIVER FUEL
B BLANKET (DEPLETED - U)
Be. Sb BERYLIUM - ANTIMONY SOURCE
C-# CONTROL ROD
S-# SAFETY ROD

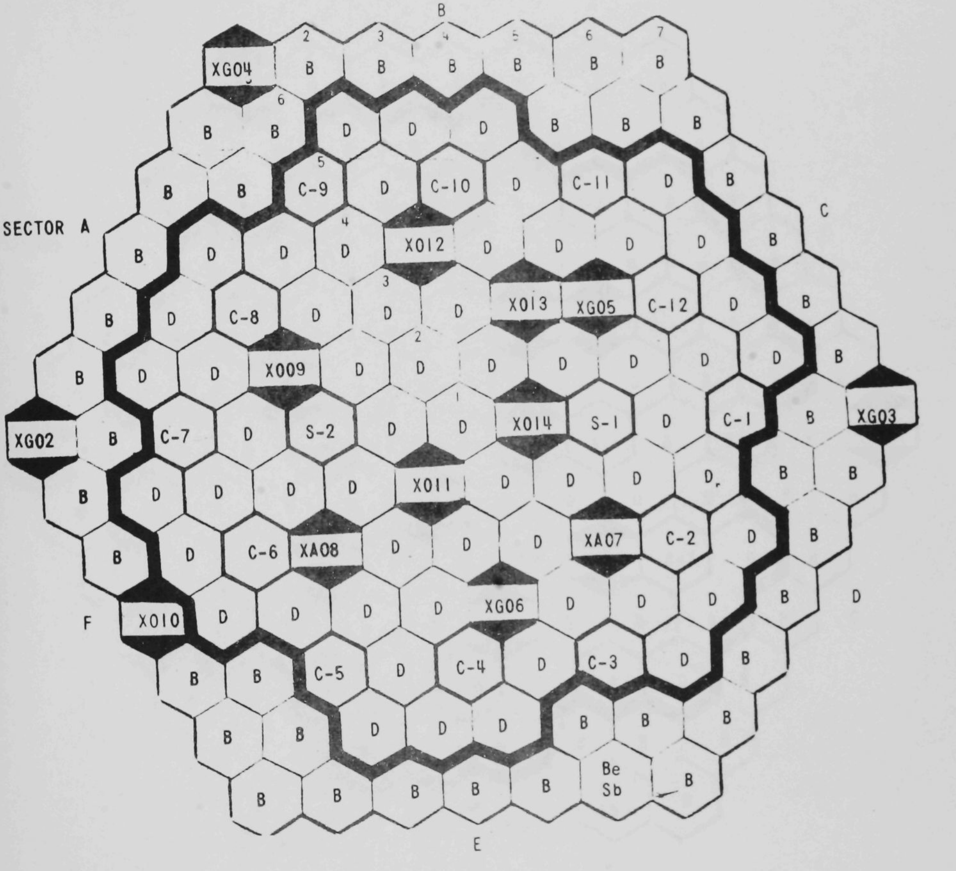


FIG. 27
EBR-II LOADING PATTERN
RUN #21

NOTE: CONTROL ROD #9 (C-9) CONTAINED STAINLESS STEEL PINS ONLY

KEY: D DRIVER FUEL
B BLANKET (DEPLETED - U)
Be. Sb BERYLIUM - ANTIMONY SOURCE
C- CONTROL ROD
S- SAFETY ROD

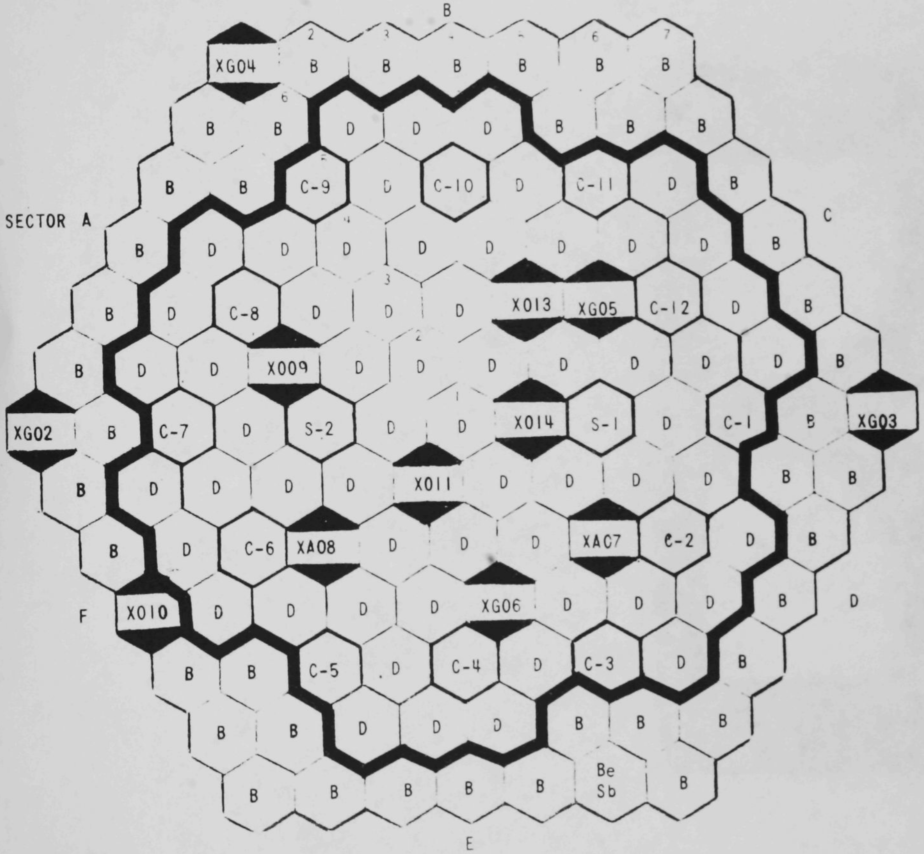


FIG. 26
EBR-II LOADING PATTERN
RUN #20